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FOREWORD

Massimo Salvatores

GIF Policy Director

Generation IV International Forum has been a challenging initiative in 2001 and a precursor of what we call today “Nuclear Renaissance”. Today it is internationally recognized that the Generation IV International Forum has been a continuous, effective and very successful focal point for collaborative R&D activities for future nuclear systems.

After the establishment of the initial Roadmap, the System Steering Committees with the help of the Expert Group have been able to launch significant collaborative projects, consistent with medium and long term objectives.

The Methodology Working Groups have produced documents that have and will have impact on future developments in key areas of future system assessment.

Most of the work performed under GIF, has led to individual technical presentations or invited presentations at International Conferences. Recently, Annual Reports have been compiled and widely distributed.

However, to foster the visibility of the overall consistency and progress of the technical work performed under GIF, it seemed appropriate to hold a *GIF Symposium*, open to the wider Generation IV scientific and industrial community.

The objective is to give a well documented state of the art of the initiative and to report and discuss the most significant technical progress and evolution in the different areas during these last ten years.

Another significant objective for this Symposium is to provide a forum for an open and hopefully lively discussion of the perspectives, priorities and challenges for the next few years, accounting for a rapidly evolving environment.

As for the Symposium program, we have organized three technical sessions and, also as part of the Symposium, a plenary session of GLOBAL 09 is fully devoted to Generation IV. We are grateful to the GLOBAL 09 Organisers to have accepted our proposal that had the objective to expose an even larger scientific community to the status and perspectives of the Generation IV initiative.

A Symposium like this one can only be successful if there is a group of dedicated, enthusiastic people that make things happen in practice. And we had the chance to be able to rely on Sunil Félix, Caroline Thoorens and on the secretariat staff at the OECD NEA, and in particular Angélique Servin and Evelyne Bertel: all of them have made an outstanding job.

Moreover, the Symposium did put an extra burden on key technical people in order to present the GIF activities and to document these activities in these proceedings to be widely distributed thanks again to the support of NEA. All of them have to be thanked warmly.

Finally, I like to acknowledge the continuous support and crucial advice of the GIF Chairman Jacques Bouchard and of all the Policy Group.

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Jacques BOUCHARD

Chairman: Generation IV International Forum

Rising energy demand, effects of global warming, and volatile prices of natural resources, are issues which largely shape today’s world economy.

In what appears as growing awareness of this twenty-first century challenge, governments in increasing numbers throughout the planet are embracing nuclear energy as vital to their strategies of national energy independence and global environmental duties.

Sustainable development of nuclear energy

However, even if nuclear energy is increasingly recognized as indispensable, it is important to ensure its sustainable use and development.

Such is the aim of the Generation IV International Forum (GIF), for which sustainability goals are defined according to criteria linked to safety, economics, resource utilization, waste management, proliferation resistance and physical protection, as well as the use of nuclear energy to applications wider than electricity production.

While the safety levels and economic competitiveness of Generation IV systems will be targeted to be at least as good as those of Generation III plants, Generation IV, as described in the 2002 Technology Roadmap, improves upon current reactors in several ways. Four of the designs are fast reactors, allowing the reactors to potentially exploit the full energy potential of uranium both fissile and fertile isotopes. Generation III reactors extract energy from a much smaller fraction of fissile uranium in the fuel, where Generation IV reactors can extend the uranium resource by about a factor of 50 beyond this. Another benefit for Generation IV is to improve on current designs by recycling all actinides not only the bred plutonium-239, but the other actinides found in the waste as well. This revolution in fuel utilization would not only dramatically reduce the long-term radiotoxicity and heat generated by the waste by transmuting it to shorter-lived fission products, thus making it easier to dispose, but also enhance the system’s resistance to proliferation, by rendering the fuel more difficult to handle.

Two of the Generation IV designs are high-temperature reactors, which can generate not only electricity but also provide high-quality process heat for industrial purposes. Process heat is useful in a wide range of industries, from petroleum refineries and chemical plants to large-scale hydrogen production that could revolutionize transportation.

The GIF: a forum for multilateral R&D cooperation

Ever since the launch of the GIF, about a decade ago, its member countries have met regularly, to discuss the research required to support the development of next-generation reactors: it has resulted in a tremendous brainstorming effort, from R&D teams from over twelve countries and EURATOM, on a scale rarely matched in history, which, in turn, produced numerous collaborative projects in reactor and fuel technologies. Even if the six nuclear systems studied within the GIF correspond to concepts already known, their development within the GIF has benefited from the exchanges between technical experts representing most of the main world nuclear actors, originating from both academia as well as industry sectors.

Though technical exchanges started long ago, especially on a bilateral level, multilateral cooperation was given a clear boost after negotiations led in 2006, to principles, accepted by all, which duly recognize background property information and deal satisfactorily with all property rights (intellectual, commercial…). The GIF thus appears as the only existing large scale international structure enabling multilateral cooperation within a sound legal basis that ensures that its R&D activities are carried out in an equitable manner between partners.

This major step was followed by the signing of a series of Project Arrangements, for most of the six GIF nuclear systems, such as, for example, that on Advanced fuel, Global Actinide Cycle International Demonstration for the Sodium-Cooled Fast Reactor (SFR) concept, Hydrogen Production, Fuel and Fuel Cycle for the Very High Temperature Reactor (VHTR) concept… Additional projects are to be finalized soon on the Gas-Cooled Fast Reactor (GFR), Super-Critical Water Reactor (SCWR) systems.

Scope of the GIF

It is very important to reiterate the scope of the work carried out within the GIF. The R&D performed focuses on both the viability and performance phases of system development: the former phase examines the feasibility of key technologies, such as, for example, adequate corrosion resistance in lead alloys or supercritical water, fission product retention at high temperature for particle fuel in the very high temperature gas cooled reactor… The latter phase focuses on the development of performance data and optimization of the system.

Conversely, the scope of GIF activities does not extend to demonstration phase, which involves the detailed design, licensing, construction and operation of a prototype or demonstration system in partnership with industry.

To help prepare for future commercial-lization of Generation IV systems, a Senior Industry Advisory Panel (SIAP) provides advice on GIF R&D priorities and strategies. Specifically, this panel contributes to discussion on strategic review of R&D progress and plans for the GIF systems from the industry perspective, by addressing issues such as industrial interest, technical viability, economics, licensing, risk management, project management and industrial infrastructure. The SIAP contributes valuable views on system deployment, future nuclear fuel cycles, and international frameworks for nuclear safety standards and regulations.

Concerning safety issues, a specific Working Group has been formed to promote a consistent approach on safety, risk and regulatory issues among Generation IV systems. More specifically, Generation IV safety goals and evaluation methodologies are being developed for use in system design and for guiding R&D plans. Interaction with the nuclear safety regulatory community, the IAEA and relevant stakeholders is also organised.

Experimental Reactor or Prototype

Though commercial deployment of Generation IV reactors cannot be foreseen before the 2030s, studies are already advanced enough for a few of the six Generation IV concepts, to start planning for the construction of experimental reactors or prototypes. The two such concepts are namely the fast spectrum ones, using gas or sodium as coolants. In the case of gas-coolant, studies underway in Europe may stimulate sufficient common interest that would in turn lead to the construction of an experimental reactor. As for sodium-coolant, it clearly appears that it corresponds to the most mature Generation IV technology. Even though final technical choices and policy decisions have not been made yet, SFR technology is a strong candidate for the construction plans of Generation IV fast reactor prototypes in France, Japan, and possibly other countries… A trilateral collaborative project was launched in 2007 by France, Japan and the US, which seeks to facilitate the commercial deployment of SFR technology by 2040 through cooperative research, shared infrastructure and joint prototype development. This project will heavily rely on the R&D results obtained within the framework of the GIF.

Interaction with other initiatives

Synergies exist and are currently being examined between the GIF and the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), mainly in the fields of safety and non proliferation. In these areas, the GIF and INPRO can be mutually complementary, the former being an R&D framework, while the latter brings together technology holders and users in view of joined actions for achieving desired innovations in nuclear reactors and fuel cycles. As for the US-led Global Nuclear Energy Partnership initiative (GNEP), the GIF is prepared to examine the possibility of providing it with the required R&D to develop, demonstrate and deploy advanced fast reactors, and their fuel cycles.

Conclusion

The road to commercial deployment of Generation IV systems still appears to be long, and paved with numerous technological challenges, which will require major breakthroughs. This situation motivates timely joint efforts by GIF Members. GIF has demonstrated political willingness to support and promote the development of sustainable nuclear energy systems, as well as enthusiasm on the part of the researchers involved in the Forum's collaborative R&D to surmount these challenges. Already, major achievements have been accomplished within the GIF's Project Arrangements, which now allow us to set some major milestones for what will be accomplished by the Forum over the next five years.

OVERVIEW OF GENERATION IV LIQUID METAL-COOLED FAST REACTORS: SODIUM-COOLED FAST REACTOR (SFR) AND LEAD-COOLED FAST REACTOR (LFR)

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I. INTRODUCTION

Sodium-cooled Fast Reactor (SFR) systems and Lead-cooled Fast Reactor (LFR) systems are among the six systems selected for joint development by the Generation IV International Forum (GIF) based on their potential to meet the GIF technology goals.[1] Both reactor types enhance sustainability by means of their fast neutron spectrum and closed fuel cycle, which serve to minimize waste and enhance resource utilization. They also have excellent potential to achieve the goals of safety and reliability, economics, and proliferation resistance and physical protection. The primary missions for both systems are electricity generation and "actinide management" (fissile consumption, conservation or breeding). Further, hydrogen production is feasible with electrolytic processes and thermochemical cycles tailored to the respective coolant temperatures.

Owing to the significant past experience accumulated with SFRs in several countries, the start-up of a prototype Generation IV SFR system is targeted for 2020.[2] The operation of a LFR Technology Pilot Plant (TPP) is also envisioned around 2020.[2]

Liquid metal reactors are designed for high-power density taking advantage of the high heat removal and high heat transport capability of the coolant.

The sodium reactor technology is comparatively mature but remains to be commercialized successfully. Drawbacks of sodium as a coolant include its chemical reactivity and opacity. Lead cooled systems are comparatively less mature but provide advantages stemming from the relative inertness and high boiling temperature of lead coolant. Drawbacks of lead coolant include its high density, corrosive nature, high melting point, and opacity. For both SFR and LFR systems, R&D is required to take advantage of their strengths and minimize their drawbacks. For example, R&D on in-service inspection and repair (ISI&R) technology is needed to assure the safety of reactor operation, in view of the opacity of sodium and lead coolants. The Generation IV International Forum (GIF) provides an effective mechanism for joint R&D in this key area and others. With distilled knowledge, information, experiences, funds and resources in the whole world to one point through GIF cooperation, based on the common Technology Roadmap, the R&D for Generation IV nuclear systems directing a unified aim is accelerated.

This paper provides an overview of the SFR systems which are formally undergoing development through a GIF System Arrangement, and the LFR systems proposed for joint development in the GIF framework; a formal System Arrangement for the LFR remains to be established.

II. SODIUM-COOLED FAST REACTORS (SFR)

II.A. Features and design options of SFRs

In several countries, experimental and prototype SFRs have been constructed and operated for more than 30 years. The SFR system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. A full actinide recycle fuel cycle is envisioned. In the Technology Roadmap for Generation IV Nuclear Energy Systems,[1] the SFR was primarily envisioned for missions in electricity and actinide management. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam and carbon-dioxide (CO_2) are considered as working fluids for the power conversion system to achieve high level performance on thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR can be competitive in electricity markets.

The three options, shown in Figures 1, 2 and 3 displaying loop-type, pool-type and modular pool-type systems, respectively, are under consideration:

- A medium to large size (600 to 1 500 MWe) loop-type SFR with MA-bearing mixed uranium-plutonium oxide (MOX) fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.[3,4]
- A medium size pool-type SFR with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities co-located with the reactor.[5]
- A small size (50 to 150 MWe) modular pool-type SFR with similar metal alloy fuel, supported by a fuel cycle based on pyroprocessing at a central or regional location.[6]

The design and performance parameters of the three options are shown in Table 1.

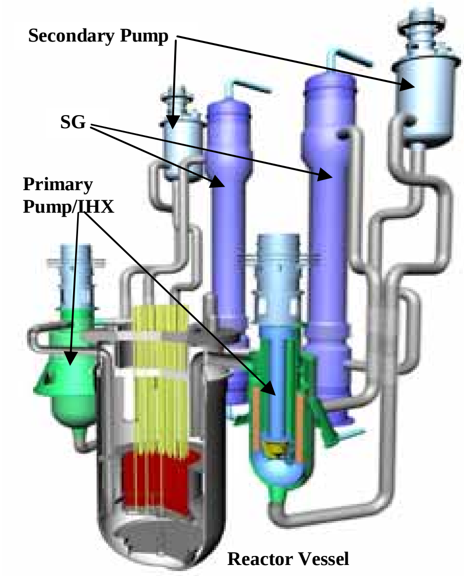

Figure 1: Loop-type SFR

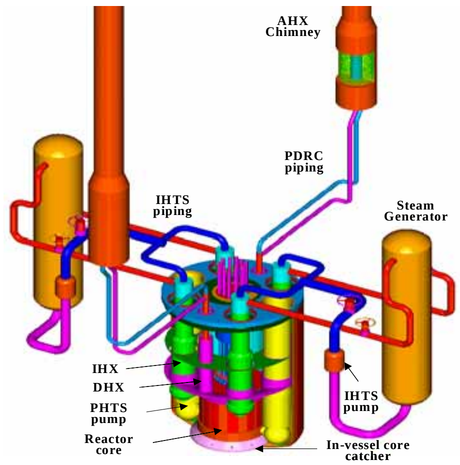

Figure 2: Pool-type SFR

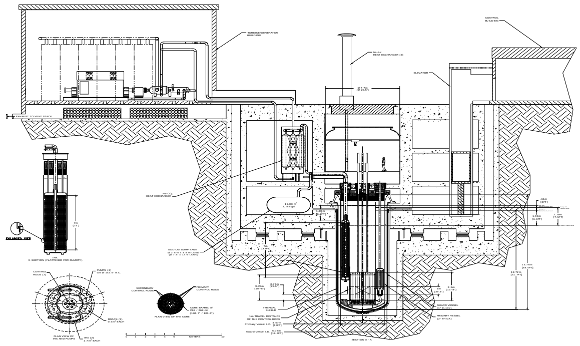

Figure 3: Small Modular pool-type SFR

Table 1: Key Design Parameters of GIF SFR Concepts

SFR Design Parameters	**Loop**	**Pool**	**Small Modular Pool**
Power Rating, MWe	1500	600	50
Thermal Power, MWth	3570	1525	125
Plant Efficiency, %	42	42	~38
Core outlet coolant temperature, ^oC	550	545	~510
Core inlet coolant temperature, ^oC	395	370	~355
Main steam temperature, ^oC	503	495	480
Main steam Pressure, MPa	16.7	16.5	20
Cycle length, years	1.5-2.2	1.5	30
Fuel reload batch, batches	4	4	1
Core Diameter, m	5.1	3.5	1.75
Core Height, m	1.0	0.8	1.0
Fuel Type	MOX (TRU bearing)	Metal (U-TRU-10%Zr Alloy)	Metal (U-TRU-10%Zr Alloy)
Cladding Material	ODS	HT9M	HT9
Pu enrichment (Pu/HM), %	13.8	24.9	15.0
Burn-up, GWd/t	150	79	~87
Breeding ratio	1.0–1.2	1.0	1.0

II.B. SFR Concepts

LOOP-TYPE SFR

To promote favorable economies of scale, many SFR designs have targeted large monolithic plant designs. For this approach, a prominent recent concept is the Japanese Sodium Fast Reactor (JSFR)[3,4] which is an advanced sodium-cooled loop-type reactor evolved from Japanese fast reactor technologies; the conceptual plant design is shown in Figure 1.

The JSFR design employs several advanced technologies to reduce the construction cost: compact design of reactor structure, shortened piping layout, reduction of loop number, integration of components, and simplification of decay heat removal system through enhancement of natural circulation capability. JSFR employs innovative technologies such as modified 9Cr-1Mo steel with high strength, an advanced structural design standard at elevated temperature, two-dimensional seismic isolation, and re-criticality free core to exclude power excursion sequences.

The JSFR design utilizes passive safety features to increase safety assurance. The improvement of the ISI&R technology is a key objective to confirm the integrity of internal structures including core support structure and coolant boundaries. The means of access is taken into account in design, and remote handling and sensor technology for use under sodium as well as a high reliable double-wall-tube SG are being developed.

While focusing on a large monolithic concept, the JSFR design studies consider plant sizes ranging from a modular system composed of medium size reactors to a large monolithic size. The large-scale sodium-cooled reactor utilizes the advantage of “economy of scale” by setting the electricity output of 1 500 MWe. On the other hand, a medium-scale modular reactor would offer advantages of flexibility in meeting power requirements from generating companies and the reduction of investment risk compared with large-scale reactors.

POOL-TYPE SFR

Moderate size SFR designs have also been proposed; in this case, cost reduction relies on design simplification and factory fabrication techniques. A recent example is the KALIMER-600(5) pool-type reactor design, shown in Figure 2, evolved from previous pool-type SFR designs such as the Power Reactor Inherently Safe Module (PRISM), the Super-Phénix (SPX) and the European Fast Reactor (EFR). A pool-type reactor provides many important design advantages in plant economy and safety. The entire Primary Heat Transport System (PHTS) piping and equipment are located inside the vessel completely eliminating the possibility of PHTS piping break outside the reactor vessel. Also the large thermal inertia characteristic of a pool-type reactor enhances passive safety mechanisms. The safety of KALIMER is enhanced further by loading its core with metal fuel which has inherent safety characteristics resulting from large negative power reactivity coefficients and a very low probability of a Core Disruptive Accident (CDA).

For improvement of the plant economy over previous designs, KALIMER reduces the number and/or eliminates equipment by design simplification and novelty, compact design and higher plant efficiency. Its net plant efficiency is designed to reach 39.3% with conventional steam plant. The introduction of the innovative Passive Decay heat Removal Circuit (PDRC) system could enable an increase in the size of the system to 1 000 MWe or more. KALIMER requires neither active-component (equipment) operation nor operator action in managing accidents. Also it does not require a safety grade emergency electricity generator. These safety design features provide very high reliability in the safety management and can accommodate design-basis events (DBE) and beyond-design basis events such as anticipated transients without scram (ATWS) without any operator action or support of active shutdown system operation. The grace period during accidents can be measured in days without violating core protection limits.

SMALL MODULAR POOL-TYPE SFR

The Small Modular Pool-type SFR (SMFR) is aimed at exploiting characteristics inherent to fast reactors for application to small grid applications. In a recent study in the United States,[6] a reactor size of 50 MWe was selected as shown in Figure 3 for a specific niche market where industrial infrastructure is not sufficient for larger systems and the unit cost of electricity generation is very high with conventional technologies.

Innovative design features have been embodied in the SMFR design including a metallic fuel core with high internal conversion ratio, passive safety characteristics, simplified reactor configuration for modular construction and transportability, and supercritical CO_2 Brayton cycle power conversion system. The primary system is configured as a typical pool arrangement and the intermediate sodium exits the vessel and flows to the sodium-to-CO_2 heat exchangers.

A key design feature of the SMFR is its long-lived core – 30 years with no refuelling. This long lifetime improves proliferation resistance by eliminating on-site fuel storage facilities and limiting fuel management to the initial insertion and eventual removal of the core. The SMFR incorporates all the passive safety features developed for SFR applications to avoid plant damage; this includes a passive decay heat removal system directly from the primary coolant pool.

II.C. Status of cooperation

The System Arrangement for the international research and development of the SFR system was signed in November 2006 by EURATOM, France, Japan, the Republic of Korea and the United States. In addition, China signed it in March 2009. Four Project Arrangements on Advanced Fuels (AF), Global Actinide Cycle International Demonstration (GACID), Component Design and Balance Of Plant (CDBOP), and Safety and Operation (SO) have been signed in 2007 for the former three and in 2009 for the SO. The Project Arrangement on System Integration and Assessment (SIA) is expected to be effective in 2009.[2]

By means of the Projects Arrangements mentioned above, the R&D activities currently being conducted are as follows:

AF: Performance evaluations for oxide, metal, nitride, carbide and nitride/ carbide fuels, MA-bearing fuel fabrication technology, and core materials for high burn-up fuels.

CDBOP: Experimental and analytical evaluation of advanced ISI&R technology including leak-before-break assessment, and alternative energy conversion system with supercritical-CO_2 Brayton cycle.

GACID: Evaluation of MA-bearing fuel material properties, analysis and evaluation of irradiated fuel data, and program planning for bundle-scale MA-bearing fuel assembly irradiation demonstration in the Monju reactor in Japan.

SO: Analyses and experiments that support safety approaches and validate specific safety features, development of computational tools useful for such studies, and acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants.

III. LEAD-COOLED FAST REACTORS (LFR)

III.A. Features and design options of LFRs

The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides from reprocessed spent fuel. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant. In the Roadmap, the LFR was primarily envisioned for missions in electricity and hydrogen production, and actinide management.

To acquire a larger experience in handling lead and resolve corrosion issues of structural material during the high-temperature operation of LFR, particularly by means of more effective and reliable oxidized surfaces, the international R&D collaboration in GIF is expected to be formalized in the future by means of a System Arrangement.

The designs that are currently proposed as candidates for international cooperation and joint development in the GIF framework are two pool-type reactors shown in Figures 4 and 5:

- the Small Secure Transportable Autonomous Reactor (SSTAR) with mixed uranium-plutonium nitride (MN) fuel.[7]
- the European Lead-cooled System (ELSY) with MOX fuel.[8]

Key design data of SSTAR and ELSY are presented in Table 2.

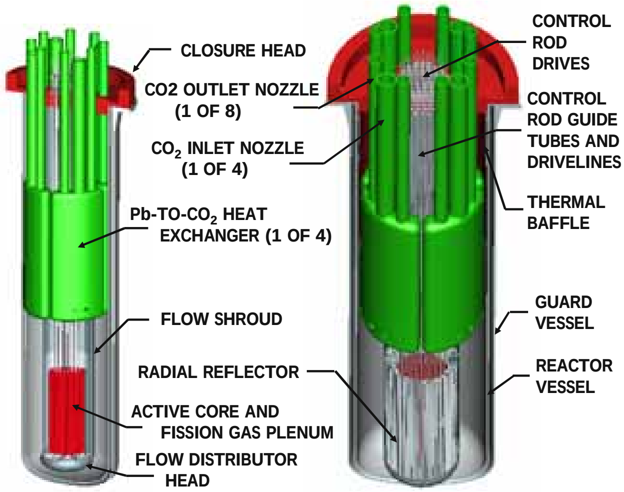

Figure 4: Small Secure Transportable Autonomous Reactor (SSTAR).

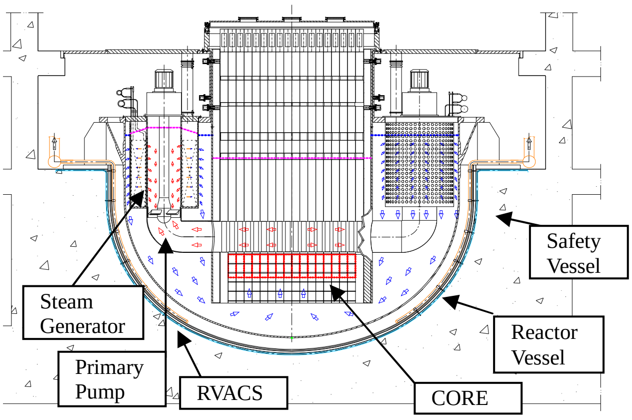

Figure 5: ELSY Primary system configuration.

Parameters/System	SSTAR	ELSY
Power (MWe)	19.8	600
Conversion Ratio	~1	~1
Thermal efficiency (%)	44	42
Primary coolant	Lead	Lead
Primary coolant circulation (at power)	Natural	Forced
Primary coolant circulation for DHR	Natural	Natural
Core inlet temperature (°C)	420	400
Core outlet temperature (°C)	567	480
Fuel	Nitrides	MOX, (Nitrides)
Fuel cladding material	Si-Enhanced F/M Stainless Steel	T91 (aluminized)
Peak cladding temperature (°C)	650	550
Fuel pin diameter (mm)	25	10.5
Active core Height/ equivalent diameter (m)	0.976/1.22	0.9/4.32
Primary pumps	-	No. 8, mechanical, integrated in the SG
Working fluid	Supercritical CO_2 at 20MPa, 552°C	Water-superheated steam at 18 MPa, 450°C
Primary/secondary heat transfer system	No. 4 Pb-to-CO_2 HXs	No. 8 Pb-to-H_2O SGs
Safety grade DHR	Reactor Vessel Air Cooling System + Multiple Direct Reactor Cooling Systems	Reactor Vessel Air Cooling System + Four Direct Reactor Cooling Systems + Four Secondary Loops Systems

TABLE 2: Key Design Parameters of GIF LFR Concepts

III.B. LFR concepts

SMALL SECURE TRANSPORTABLE AUTONOMOUS REACTOR (SSTAR)

The current reference concept for the SSTAR[7] in the United States is a 20 MWe natural circulation reactor concept with a small shippable reactor vessel, as shown in Figure 4.

The lead coolant is contained inside a reactor vessel surrounded by a guard vessel. Lead is chosen as the coolant rather than lead-bismuth eutectic (LBE) to avoid generation of alpha-emitting ^{210}Po via neutron interactions with bismuth, and to eliminate dependency upon bismuth which might be a limited or expensive resource. The lead flows upward through the core and a chimney above the core formed by a cylindrical shroud. The coolant enters four modular lead-to-CO_2 heat exchangers located in the annulus between the reactor vessel and the cylindrical shroud.

Physical properties of the lead coolant, the nitride fuel containing transuranic elements, the fast spectrum core, and the small size combine to promote a unique approach to achieve proliferation resistance, while also enabling fissile self-sufficiency, autonomous load following, simplicity of operation, reliability, transportability, as well as a high degree of passive safety. Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished utilizing a supercritical CO_2 Brayton cycle power converter.

EUROPEAN LEAD-COOLED SYSTEM (ELSY)

The ELSY[8] reference design is a 600 MWe reactor cooled by lead, shown in Figure 5. ELSY has been under development since September 2006, and is funded also by EURATOM within the Sixth Framework Programme. The ELSY project is being performed by a large consortium of European organizations to demonstrate the possibility of designing a competitive and safe fast critical reactor using simple engineered features, while satisfying Generation IV goals, including waste minimization and effective waste management through minor actinide consump–tion (burning).

Simplicity and reduced footprint would be possible due to the elimination of the intermediate cooling system and the identify-cation of innovative solutions to reduce the primary system volume and the complexity of the reactor internals.

ELSY features a cylindrical inner vessel, axial-flow primary pumps located inside the inner shell of the spiral-tube bundle SGs, and safety decay heat removal system with lead-water dip coolers. Because fuel assemblies are largely sustained by buoyancy and kept in the vertical position by support beams in gas space, the hitherto classical core support plate has become needless, and the refuelling machine can operate in gas instead of in lead.

All reactor internal structures are removable; particularly the SG Unit can be lifted off by radial and vertical displacements which disengage the unit from the reactor roof. Above-mentioned technologies would contribute greatly to reducing necessity of ISI&R in molten lead.

The core consists of an array of open (wrapperless) fuel assemblies (FAs) of square pitch surrounded by reflector-assemblies, a configuration that presents reduced risk of coolant flow blockage. An alternative solution with closed hexagonal FAs has been retained as a fall-back option.

III.C. Status of cooperation

Preparation of a System Arrangement for approval by participating GIF members has been considered, but formal agreements are still pending. The LFR System Research Plan (SRP) is under preparation by the Provisional System Steering Committee (LFR-PSSC) with the participation of the Representatives from EURATOM, Japan, the United States and experts from the Republic of Korea. In addition, informal meetings were held with the participation of the representatives of the nuclear industry, research organizations and universities involved in LFR development.[2]

VI. CONCLUSION

Some candidate concepts of SFR and LFR systems, which take advantages of respective features, are proposed. GIF participants are going to conduct cooperative R&D concluding project arrangements around common techniques to each system.

Profiting from the experience of the R&D and operation of the experimental and prototype SFRs acquired over many years, the international collaborative R&D activities for the SFR system within GIF are being successfully conducted; start-up of a prototype system is targeted for 2020.

The LFR system is proposed for joint development within GIF. The draft LFR SRP describes a dual track viability research program with convergence to a single, combined Technology Pilot Plant (TPP) leading eventually to the deployment of both types of systems (SSTAR and ELSY). Following the successful operation of the TPP around the year 2020, an independent development of two prototypes is expected to lead to a subsequent industrial deployment of the central station LFR and the SSTAR, respectively.

References

1. The U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, “A Technology Roadmap for Generation IV Nuclear Energy Systems”, December 2002.
2. Generation IV International Forum, “2008 Annual Report”.
3. Kotake, S., *et al*., “Feasibility Study on Commercialized Fast Reactor Cycle Systems/Current Status of the FR System Design”, Proceedings of Global 2005, Tsukuba, Japan, 9-13 October 2005.
4. Mizuno, T., *et al*., “Advanced Oxide Fuel Core Design Study for SFR in the Feasibility Study in Japan”, Proceedings of Global 2005, Tsukuba, Japan, 9-13 October 2005.
5. Hahn, D., *et al*., “Design Concept of KALIMER-600”, Proceedings of Global 2005, Tsukuba, Japan, 9–13 October 2005.

6. Chang, Y., *et al*., "A Case for Small Modular Fast Reactor", Proceedings of Global 2005, Tsukuba, Japan, 9-13 October 2005.
7. Sienicki, J.J. *et al*., "Status of development of the Small Secure Transportable Autonomous reactor (SSTAR) for Worldwide Sustainable Nuclear Energy Supply", Paper 7218, Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP), Nice, France, 13-18 May, 2007.
8. Cinotti, L., *et al*., "The ELSY Project", Paper 377, Proceeding of the International Conference on the Physics of Reactors (PHYSOR), Interlaken, Switzerland, 14-19 September, 2008.

High Temperature Reactors (VHTR & GFR)

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I. INTRODUCTION

Assets of high temperature reactors include both potentialities of producing electricity with high conversion efficiency and supplying process heat above 600°C which put them in a position to supplement light water reactors for displacing fossil fuels in a wide range of applications. They may indeed cogenerate steam, hydrogen and heat for varied industrial sectors such as oil industry (refinery, synthetic transportation fuels…), chemistry and steelmaking. Owing to growing concerns about climate change high temperature reactors (HTRs) that led to prototype reactors from the 1960s through the 1980s currently experience a revival of interest in the form of new projects of reactors by 2015-25 and cooperative R&D for a *Very High Temperature Reactor (VHTR)* that materializes a long term vision of this reactor type in the Generation IV International Forum. New builds are planned in China, South-Africa and the United-States with the aim of testing modern technologies for high temperature reactors and demonstrating non conventional nuclear applications at pre-industrial stage. Multinational cooperation in the Forum complements national R&D efforts for these projects of reactor at 700-850°C and also develops technology break-throughs for the VHTR aiming at 900-1 000°C. All active members of the Forum currently contribute to the development of VHTR technologies. The renewed interest in the *Gas-cooled Fast Reactor (GFR)* stems from its dual assets of being both an alternative type of fast neutron reactor avoiding critical issues associated with liquid metals, and a vision of highly sustainable high temperature reactor enabling a durable production of varied energy products. Both types of gas-cooled reactors complement each others as the VHTR is a stepping stone towards the GFR which in turn opens up more durable prospects for VHTR specific missions. GFR specific R&D that is currently conducted by five members of the Forum focuses on ceramic clad carbide fuels, design studies and safety analyzes, and results obtained so far have established confidence in the feasibility and potential performances of this type of reactor.

Both types of gas-cooled reactors call for synergistic R&D on heat resisting materials, helium systems' technology and power conversion systems which are addressed in the Forum as cross-cutting R&D projects. Pre-conceptual studies of an experimental GFR supplement current R&D work to prepare a possible decision of build in the next decade. Among the six Generation IV systems the VHTR and the GFR constitute a consistent and versatile set of reactors with high potential which arouses a growing interest as the Forum expands.

II. PAST EXPERIENCE ON HTRs AND GFRs

Five experimental and prototype high temperature reactors were built and operated from the 1960s through the 1980s in the United-States and Europe with block-type and pebble bed core designs respectively: *DRAGON (20 MW_{th})* in the UK, *Peach Bottom (40 MW_e) and Fort-Saint-Vrain (330 MW_{th})* in the US, *and AVR*

High Temperature Reactors (VHTR & GFR)

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(15 MW_e) and THTR (300 MW_e) in Germany. They demonstrated the technical viability of this reactor type but could not prove their economic competitiveness with light water reactors for electricity production. No further developments were to occur until the late 1990s when the interest in HTRs was revived by needs of low carbon high temperature heat supply for varied industrial processes.

Projects of gas cooled fast reactors also existed from the 1960s through the 1980s as an alternative to sodium cooled fast reactors that would avoid complex liquid metal coolant technology but no prototype was ever built as the slow development of nuclear energy postponed the need for fast neutron reactors and the safety of these early gas fast reactors was challenged by the use of conventional steel clad fuel and the desire for high power density for short doubling times. The Gas-Cooled Fast breeder Reactor program (GCFR) led by General Atomics in the United States has probably been the most active initiative in this direction. These developments came to an end in the late 1980s and were revived in 2001 with a new vision of Gas-cooled Fast Reactors (GFR) within the framework of the Generation IV International Forum.

III. TODAY'S CONTEXT

III-A – Current Experimental High Temperature Reactors

First, the Japan Atomic Energy Agency (JAEA) built a research reactor in Oarai, the High Temperature engineering Test Reactor (HTTR) that was put in service in 1998 and reached its full design power of 30 MW_{th} in 1999 with an outlet helium temperature of 850°C. Subsequent tests have demonstrated the safe behavior of the reactor in various accidental sequences and the successful operation at the design temperature of 950°C. The HTTR restarted in 2009 after 18 months at shutdown. It will proceed with a continuous operation at 950°C for 60 days. In parallel with tests on the HTTR, JAEA is developing the sulfur-iodine thermo-chemical process to produce hydrogen. A first demonstration of this process was achieved in 2003 when a continuous production of 30 litres of hydrogen per hour was obtained for a few days. The next steps are tests of a pilot plant of 400 kW (30 m^3/hr) around 2012 and tests of nuclear production coupled to the HTTR at pre-industrial scale (10 MW and 1 000 m^3/hr) around 2015-2020.

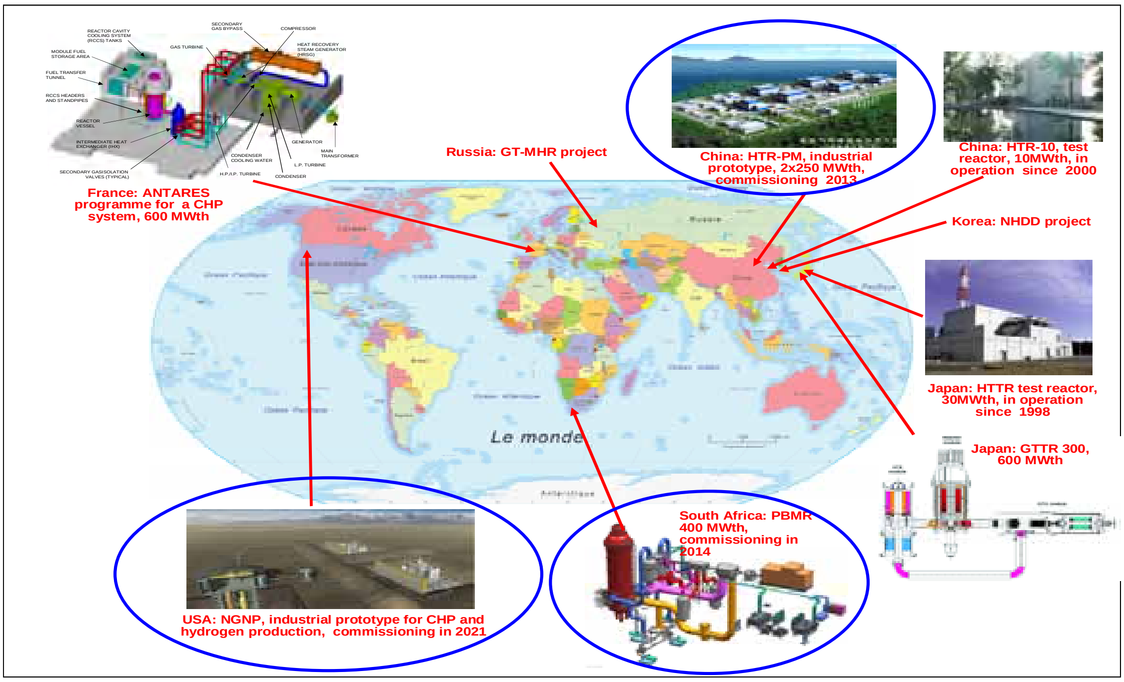

Figure 1: Status of HTR development in the world

Then, Institute of Nuclear and New Energy Technology (INET) of Tsinghua University in China built the experimental reactor HTR-10 (10 MW_{th}) that was put in operation in 2000. The successful operation of this reactor demonstrated an updated pebble bed core HTR technology and paved the way for scaling up this technology into the HTR-PM[1] project in China.

Currently, the revival of interest in high temperature process heat applications fostered R&D and projects of new builds of HTRs in the world for the period 2015-2025 (Figure 1) thus preparing the advent of a new generation of this reactor type (*the Very High Temperature Reactor (VHTR)*) and its varied applications: electricity first and process heat in a second stage, or dedication to hydrogen production

III-B – Ongoing High Temperature Reactors International Projects

The interest and support of end user industries is sought to create private/public partnerships to build and operate such prototypes and proceed with demonstrations relevant to their industrial needs. Industrial sectors concerned include the oil industry (extraction & treatment of oil sands, production of synthetic fuels from coal & biomass), as well as chemical and steel industries.

In 2005, **China** announced its intention to scale up the HTR-10 technology and to realise a national project of 200 MW_e commercial plant with independent intellectual property rights. This project consists in two High Temperature Reactor-Pebble Bed Modules (HTR-PM) [1] of 250 MW_{th} with a helium core outlet temperature of 750°C that drive together a steam turbine of 200 MW_e. The construction has begun in 2009 on the site of the Shidaowan plant in the Province of Shandong with a commissioning planned in 2013. The reactor is designed to ultimately achieve a core outlet temperature of 950°C with current core design and fuel element technologies. Besides, the modular nature of the HTR-PM makes it possible to replace the steam turbine of the power conversion system by a helium turbine or a super critical steam turbine, as well as by a hydrogen production plant in a second stage.

In the **Republic of South Africa**, the Pebble Bed Modular Reactor Pty. Ltd (PBMR) [2] is a public-private partnership that was established in 1999 to initiate the development of a modular pebble-bed reactor with a rated capacity of 165 MW_e. In 2009 the PBMR project had its business re-oriented towards the supply of industrial process heat. Thus, PBMR Ltd started developing options for commercial fleets with Sasol for producing synthetic fuels from coal, with Eskom for electricity, as well as with US and Canadian cogeneration end users including oil sand producers. The PBMR project was accordingly revisited as a cogeneration steam plant with a thermal power of 200 MW_{th}, a helium temperature of 750°C at core outlet and a steam generator directly placed in the primary loop.

In the **United States**, the *Next Generation Nuclear Plant* (*NGNP*) [3] project was mandated by the US Energy Policy Act of August 8, 2005 as a high-temperature gas-cooled reactor intended for high-efficiency electricity production, high-temperature process heat generation, and nuclear-assisted hydrogen production at the Idaho National Laboratory (INL). It would be co-located with an industrial plant that would use process heat from the reactor and could operate in 2021. Pre-conceptual and conceptual design studies concluded that there are no discriminating technical factors that favor pebble bed or prismatic design over another and that the initial gas outlet temperature will be in the 750-800°C range to meet most users' needs. The NGNP project took another step in August 2008 when the US-DOE and the NRC submitted a joint licensing plan leading to a license application filed in 2013. DOE is currently developing a final strategy for partnering with the industry (nuclear vendors and potential users of process heat in sectors such as oil-, chemistry or steelmaking) to drive the development of the NGNP project.

In **Japan**, the Japan Atomic Energy Agency (JAEA) is currently conducting research and development for the project of "*Gas Turbine High Temperature Reactor 300 – Cogeneration*" (*GTHTR300C*) [4] that is dedicated to CO_2 emission free cogeneration of electricity and hydrogen by sulfur-iodine thermo-chemical water splitting process. With a thermal power of 600 MW and a block-type core with an exit temperature of 950°C, the GTHTR300C is believed to be highly efficient and economically competitive for cogenerated hydrogen and electricity.

In the **Republic of Korea**, the context of wilful development of hydrogen technologies to prepare the hydrogen economy led the Korean Atomic Energy Commission to approve in Dec.'08 a national program on key technologies development for nuclear hydrogen and a project of Nuclear Hydrogen Development and Demonstration (NHDD) [5] This project aims at designing and constructing a nuclear hydrogen production system, as well as demonstrating its safe and reliable operation. The project is expected to be launched in 2010 with target dates of 2022 for the completion of construction and 2026 for prototypical demonstrations.

In **Europe**, a partnership of European nuclear industrial and research organisations for developing HTR technologies has been established with the creation in 2000 of the (European) "HTR Technology Network" (HTR-TN). HTR-TN has played since then a prominent role in defining a strategy for European R&D on HTRs and implementing this strategy in Euratom Framework Programmes (FP) since 2000 (5^{th} FP). This led to revive in the 6^{th} FP (2002-06) the past experience in Europe on HTR design tools and technologies (fuel, materials, helium systems' technology, coupling technologies…) in a program called RAPHAEL. This set the stage for Euratom to bring consistent contributions to VHTR R&D Projects in the Generation IV International Forum and for approaching industrial sectors potentially interested in low-carbon process heat. However, marketing prospects of high temperature nuclear heat are still too uncertain for stakeholders of the nuclear industry and potential users of HTR energy products to envision yet building a prototype of next generation HTR in Europe.

In parallel the 6^{th} FP also established a cooperative framework in Europe on Gas-cooled Fast Reactors through the action GCFR and its successor GoFastR in the 7^{th} FP. This created a community that supports a project of experimental prototype of *Gas Fast Reactor* as an option to be documented for decision by 2012 to advance alternative fast reactor types in parallel to a prototype of new generation sodium fast reactor planned for 2020 within a European Technology Platform on "Sustainable Nuclear Energy" (SNE-TP) [6] launched in 2007.

IV. CURRENT GIF ACTIVITIES ON VERY HIGH TEMPERATURE REACTORS (VHTRs)

The potential of a VHTR at 900-1 000°C to match temperature requirements for advanced hydrogen production processes based on electro- or thermo-chemical water splitting processes was the initial driver for selecting this reactor type in 2002 among the six Generation IV Systems. Missions of the VHTR have expanded since then to cogeneration of electricity and process heat for varied industrial applications. [7] This system experiences a sustained interest from all active members of the GIF since its beginning. The VHTR System Arrangement was signed in December 2006 by Canada, Euratom, France, Japan, the Republic of Korea, Switzerland and the United-States. The People's Republic of China signed this Arrangement in October 2008. Multinational cooperation in the GIF complements national R&D efforts for current projects of reactor at 700-850°C and also develops technology breakthroughs for the VHTR aiming at 900-1 000°C. R&D Projects on "Fuel and fuel cycle" and "Hydrogen production" became effective in January and March 2008 and a project on "Materials" has become effective in the fall of 2009. A project on "Computational methods, validation and benchmarking" will be ready for signature in early 2010.

Specific Agreements will be worked out to frame exchanges between cooperative R&D in the GIF and VHTR related projects so as to assure a fair treatment of R&D results generated by GIF members and their privileged access to operating parameters of prototype reactors in fair conditions.

IV-A – Fuel Fabrication and Qualification

Cooperative work on TRISO fuel includes sharing irradiation experiments, characterization methods and facilities as well as constituent materials properties. Besides, research is also conducted on advanced fuel particles such as UCO fuel and ZrC coated particles. GIF contributes to sharing the effort to reacquire the mastery of standard TRISO coated fuel particles fabrication and qualification. Figure 2 shows the laboratory-scale line CAPRI that was put in operation in 2005 on the CEA site of Cadarache as part of the effort to revive HTR technologies in France.

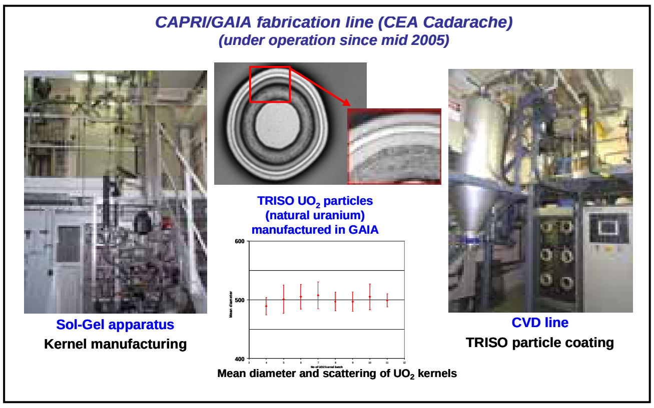

Figure 2: CAPRI/GAÏA fabrication line of fuel particles (CEA-Cadarache)

The cooperation within the GIF takes an especially active part in sharing irradiation services for TRISO fuel. Within this framework, an invitation was extended by the United States to France and the Republic of South Africa to join with their own samples the American TRISO particles test program AGR2 in the Advanced Test Reactor (ATR) that will begin in early 2010. The cooperation is also active on characterization methods to check and improve the fabrication quality of TRISO particles first at laboratory-scale and then at industrial scale.

The management of spent TRISO fuel particles and that of used graphite are also parts of the cooperative work within the GIF. Acoustic waves and pulsed currents de-structuring methods are tested on dummy coated particles to retrieve nuclear materials from the kernel for recycling and package coatings as ultimate waste. The same de-structuring methods are also tested on graphite as a first step for processing used graphite and partitioning ^{14}C for disposal. These issues are also addressed at the European level in the 7th Framework Program in a research program (CARBOWASTE) dedicated to best practices for retrieval, treatment and disposal of used graphite and other carbonaceous forms.

IV-B – Materials and Components

Cooperative development of materials covers graphite, advanced super-alloys (nickel-based and 9Cr ferritic steels) and composite ceramics. It aims at screening and qualifying structural materials for key components of VHTRs and particularly high temperature heat exchangers. The experimental work is defined in common and shared among participating GIF members. Mechanical and corrosion tests are conducted to screen candidate materials and to acquire data needed for extending current design codification rules in VHTR service conditions and for licensing prototype reactors of this type. Results are compiled in a common data base operated by the Oak Ridge National Laboratory.

The 9Cr1Mo alloy is currently characterized as promising candidate for a hot reactor pressure vessel operating at 400-450°C (*i.e.* beyond limits of the SA 508 steel commonly used in PWRs). Besides, two conventional nickel-base alloys (617 and 230) are currently characterized at temperatures ranging from 700°C to 1 000°C in terms of mechanical properties and corrosion resistance for use as structural material, especially for the intermediate heat exchanger.

An example of shared experimental work that consisted in coordinating irradiation tests of various grades of graphite at different temperatures is shown on Figure 3. It contributed to significantly accelerate the characterization of candidate grades of graphite in relevant service conditions for use in block-type or pebble-type HTR cores.

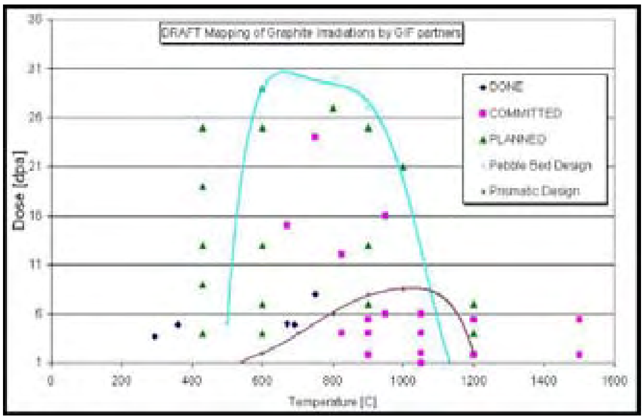

Figure 3: Test matrix (dose, temperature) of graphite samples for use in block-type or pebble type HTR cores

Components such as compact heat exchangers, as well as associated manufacturing technologies and tests of mock-ups on helium loops are currently conducted on a national basis and have not given rise to cooperative work so far.

IV-C – Hydrogen Production

Cooperation on hydrogen production processes includes:

- Sharing basic R&D to establish optimized flow-sheets and update assessments of technical/economic performances
- Sharing laboratory scale experiments to demonstrate key features for the feasibility and performance of water splitting processes (high temperature steam electrolysis, sulfur-iodine thermochemical cycle, and hybrid sulfur cycle)
- Investigating and developing technologies to couple the reactor and the hydrogen production process
- Establishing common plans for next step experiments in the range of 0.5 – 1 MW as well as for pre-industrial demonstrations with the HTTR (~2015) and near term HTR projects in the 2020s.

GIF cooperative framework contributed to share the realization and results of laboratory scale experiments on the sulfur-iodine and high temperature electrolysis, to advance the development of catalysts and to share results of technical and economic assessments of varied candidate water splitting processes. An example of shared experimental work consisted in an Integrated Laboratory Scale experiment of the sulphur-iodine thermochemical process that was jointly constructed and operated by CEA, Sandia National Laboratories and General Atomics in 2007-2008 on the site of the latter in San Diego. This experiment, that was designed for a production rate of 100 l/hour confirmed the difficulty to manage iodine in chemical processes, even at laboratory scale, and contributed together with economic analyses to orient priority research of some GIF members towards high temperature steam electrolysis.

IV-D – Computer Code for Design Studies

Cooperative work on computational methods, validation and benchmarks includes:

- Sharing analyses of key aspects of VHTR designs that call for priority improvements in modelling and simulation methods (PIRT analysis)
- Comparing computational methods for predicting VHTR key design and operating parameters
- Sharing experimental results to qualify computational methods in use for national VHTR-related projects.

V. CURRENT GIF ACTIVITIES ON GAS FAST REACTORS (GFRs)

The renewed interest in gas-cooled fast reactors stems from their potential for being both:

- A real alternative to sodium cooled systems as an attractive fast reactor concept featuring a good breeding capability and high plant efficiency owing to the low neutron absorption and high temperature capability of helium as a coolant, and

- A sustainable high temperature reactor for a durable cogeneration of non-electricity energy products.

Gas as coolant implies a poor thermal inertia and a reduced heat transfer capability, which both call for heat resisting fuel forms (ceramic clad) and redundant/diversified systems to safely manage cooling accidents. In return, helium exhibits definite advantages such as single phase cooling in all situations, chemical inertness, transparency to neutrons (hence a reactivity effect of coolant void <1$), optical transparency likely to facilitate in service inspection, maintenance and repair…

GFR studies that were launched in 2001 within the framework of the Generation IV International Forum led to defining in 2005 a ~1 100 MW_e reference concept with a core outlet temperature of 850°C as a result of active international cooperation with major inputs from both JAEA and USDOE-ANL. As planned in the GIF roadmap of the GFR, results of conceptual studies and operating transient analyzes were compiled in a pre-feasibility report [8] at the end of 2007 and presented in an international seminar hosted on February 5-6, 2008 in Paris. These results globally established confidence in the feasibility and safety of the considered GFR baseline concept. Furthermore they identified priority R&D to support further feasibility demonstrations and update the baseline concept by 2012 with innovative design features such as pin-type fuel and a pre-stressed concrete pressure vessel for improved performance. This milestone is essential as it coincides with the end of the GFR viability phase, with the issue of a final viability report, and with decisions about detailed studies for construction of a 50-100 MW_{th} experimental GFR (project “Allegro”).

V-A– Fuel and Core Design

Heat resisting fuel forms constitute the key feasibility issue of GFR’s feasibility and performance. Requirements that were considered at first to assure a safe management of most severe cooling accidents include keeping sufficient cladding integrity to contain fission products up to 1 600°C, and preserving the geometry of the fuel element up to 2 000°C. Plate or pin shaped carbide fuels with SiC-SiC_{fibers} composite cladding have been the subject of modelling and laboratory-scale R&D since 2001. The main focus was first put on plate fuel (Figure 4) that was selected as reference for the GFR baseline concept documented in 2007. Current plans include testing plate fuel to advance feasibility demonstrations of the base line concept, and shifting the main focus of R&D to pin fuel with multilayer composite cladding and compliant thermal joints between fuel pellet and cladding.

Carbide fuel (U, Pu)C or (U, Pu, MA)C with minor actinides is taken as reference fuel for its high heavy atom content and good thermal conductivity that feature excellent neutronic properties *(core critical size, breeding)* and moderate normal operating temperature (~1 100°C at 100 MW/m^3 average core power density *vs* a melting temperature above 2 350°C). Zirconium silicide (Zr_3Si_2) is identified as promising material for the neutron reflector.

The collaboration within the GIF is essential to share developmental work on advanced composite forms considered as fuel cladding and other core structural materials for the GFR. This work benefits from synergies with the development of ceramics for fusion reactors’ first wall and blanket.

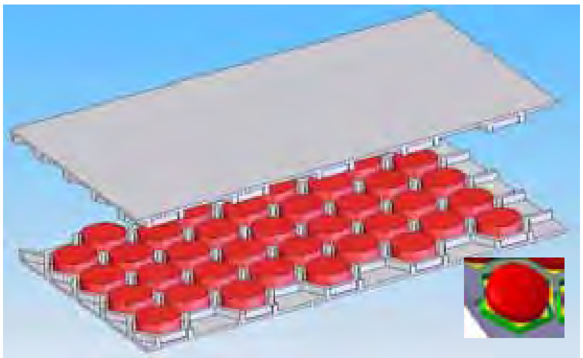

Figure 4: GFR plate fuel design, cell details (fuel pellet in red, clad in grey, leak-tight barrier in yellow/green)

Fuel plates are fitted in hexagonal wrapper tubes (as in sodium-cooled fast reactors) that constitute robust fuel subassemblies and thus assure core stability and mechanical equilibrium (Figure 5). A first design of the fuel subassembly was performed with available thermal-mechanical models so as to minimize the volume fraction of structural materials and keep acceptable stress levels.

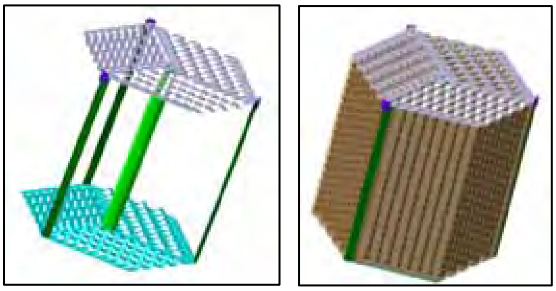

Figure 5: Basket, basket filled with plates.

A first demonstration of fuel viability will be achieved by the end of 2012 though a variety of irradiation tests including FUTURIX-MI (inert materials) and FUTURIX-CONCEPTS (fuel concepts) in PHENIX and IRRDEMO (plate and pin prototype fuel elements) in BR2 (SCK-Mol).

V-B– Core and System Design

The current core design features a plutonium hold-up of ~10 t/GW$_e$ that is comparable with the performance of sodium cooled reactors and fairly acceptable for an industrial deployment in a fleet of reactors. At equilibrium, the core achieves breeding without blankets, the fraction of minor actinides in the core reaches 1.1% and reactivity coefficients are quite satisfactory to provide sound reactivity feed-back effects for safety: the effect of helium depressurization is less than the delayed neutron fraction (<1$), the Doppler coefficient is larger than in SFRs – owing to a softer spectrum in GFRs – thus resulting in a markedly stabilizing effect.

The reference version of the Gas-cooled Fast Reactor that was selected in 2007 by GIF participants in the GFR system features a 2 400 MW$_{th}$ (~1 100 MW$_e$) reactor with three primary cooling loops (800 MW$_{th}$ each) indirectly coupled to a power conversion system using a combined cycle composed of three gas turbines and one steam turbine. In a first approach, the same reactor pressure vessel as that considered for the GT-MHR (Gas-Turbine Modular Helium-cooled reactor) was taken for the GFR assuming that associated manufacturing and other issues had been investigated and resolved (forging, welding, transport…). The primary system is illustrated on Figure 6.

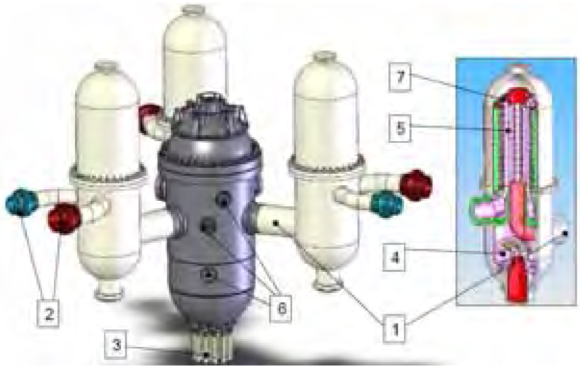

Figure 6: GFR reactor pressure vessel and IHX vessels (and exploded view of one IHX – blower unit)

<u>*Key*</u>

1. *Primary cross-duct*
2. *Secondary pipes with isolating valves*
3. *Control Rod Drive Mechanisms*
4. *Primary blower and associated motor*
5. *Compact Heat Exchanger modules*
6. *Pipe connections for Decay Heat Removal systems*
7. *Primary isolation valve*

Each of the three cooling loops is fitted with a 800 MW$_{th}$ IHX-blower unit enclosed in a single vessel (indirect cycle). Intermediate heat exchangers are mounted above the core to ease natural circulation of flow across the core. GFR cooling systems design studies strive to take maximum benefit from relevant technology developments that are currently performed for the VHTR, especially for advanced gas/gas intermediate heat exchangers. The secondary side of the IHX that is connected to the power conversion system uses a mixture of helium and nitrogen at 6.5 MPa as working fluid (Figure 7).

Electrical power is generated by the three gas turbines (3 x 130 MW$_e$) and the unique steam turbine (730 MW_e) thus achieving an efficiency close to 45% with an inlet core temperature of 400°C. This performance can be further improved while optimizing component efficiencies and pressure drops.

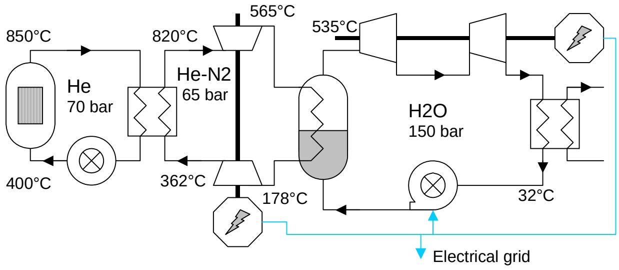

Figure 7: Indirect combined cycle – Arrangement of power conversion system

V-C– Decay Heat Removal Strategy and Safety Systems

Owing to the low thermal inertia of the GFR core, an efficient and reliable Decay Heat Removal (DHR) strategy is of utmost importance to assure a robust management of cooling accidents, including fast reactor blow-down. The cooling strategy assumed in GFR's reference-2007 design relies on assuring a gas flow across the core by forced convection in the short term and by natural convection at a back-up pressure of 0.4 – 1 MPa typically one day after the accident (and before in most accidental situations). [9, 10] The back-up pressure is maintained by a guard containment that encloses the primary system. Gas injection from pressurized nitrogen tanks complement a diversified set of cooling loops designed to operate over the whole pressure range from nominal to back-up. Figure 8 shows the lay-out of the various DHR systems (normal and back-up) as well as their integration in the guard containment vessel and the containment building.

A preliminary safety analysis of the current GFR with DHR systems design shows that decay power can be removed reliably with moderate pumping power and/or natural convection in any postulated accident including large breaks with multiple additional failures (with the addition of nitrogen injection for some of the most severe loss of coolant accidents). Typical pumping power needs (a few 100s KW_e) may be supplied by batteries.

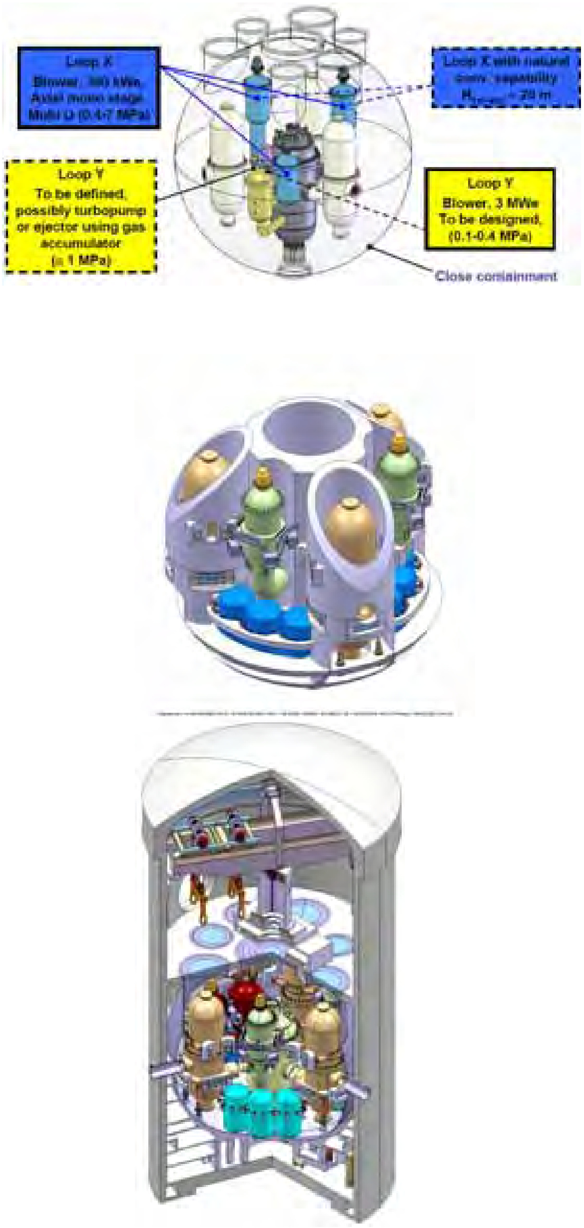

Figure 8: Schematics of varied DHR systems (normal and backup) and their integration in the guard containment vessel and containment building.

Current GFR design studies are supplemented by pre-conceptual studies of a 50-100 MW_{th} experimental facility that could demonstrate in the 2020s GFR key technologies and operating principles, and provide multipurpose fast spectrum irradiation services (project "Allegro").

VI. FUTURE PROSPECTS

As evidenced by large national projects of prototype gas-cooled reactors such as HTR-PM in China, PBMR in the Republic of South Africa and the NGNP in the United States, today's concerns of energy security and climate change open up renewed perspectives for high temperature gas-cooled reactors and for demonstrations of nuclear cogeneration of non-electricity energy products. This, together with the fact that all active GIF members contribute to R&D on VHTRs, acknowledges the potential of this reactor type to displace fossil fuels in varied applications such as producing electricity, non-conventional hydrocarbon fuels from coal or biomass, and process heat for energy intensive industries (oil refining, oil-sand recovery, petro-chemistry, chemistry, steelmaking…). Current prospects of carbon taxes and rising oil prices favour potential applications of high temperature reactors.

Besides, the support brought by a subset of GIF members to a renewed vision of the Gas-cooled Fast Reactor with very high temperature resisting fuel forms acknowledge the potential of this reactor type to be a real alternative to sodium cooled fast reactors with good performances and safety features in spite of a less efficient coolant than liquid metal. It also acknowledges the potential of this reactor type to more sustainably achieve VHTRs' high temperature applications owing to the better utilization of uranium afforded by fast neutrons.

Among the six Generation IV systems the VHTR and the GFR constitute a consistent and versatile set of reactors with high potential which arouses a growing interest as the GIF expands. The VHTR acts as a stepping stone towards the GFR (by supporting the development of cooling and conversion system technologies for both reactors), and the GFR opens up more durable prospects for VHTR specific missions.

Cooperative research projects in the Generation IV International Forum and the European Sustainable Nuclear Energy Platform supplement national programs to develop both types of gas-cooled reactors. They speed-up the development of key technologies (very high temperature technologies, refractory fuels, advanced conversion systems…), they spur the interest of process heat using industries in high temperature reactors, and they favour the creation of consortiums with the industry interested in building prototypes as public/private endeavours.

Gas-cooled reactors currently encounter less support in Europe than they do in other parts of the world and various initiatives within the European Technology Platform SNE-TP aim at strengthening research on these reactor types and fostering decisions on a VHTR prototype for demonstrations of cogeneration, as well as promoting a 50-100 MW_{th} experimental facility supporting GFR's demonstration needs and multipurpose fast spectrum irradiation services (project "Allegro").

Beyond being essential for sharing costs of R&D and prototypes, the development of strong frameworks of multilateral cooperation will also be essential to support the development of harmonized international standards that would apply to future nuclear systems in terms of safety, design rules, physical protection and non-proliferation.

Acknowledgements

Authors acknowledge the inputs from members of both GIF's VHTR and GFR Steering Committees to this review paper of international R&D on High Temperature Reactors.

Nomenclature

AEC – Atomic Energy Commission
AVR – Arbeitsgemeinschaft Versuch Reaktor
CEA – Commissariat à l'énergie atomique
FP6, FP7 – 6th, 7th European R&D Framework Programme
GIF – Generation IV International Forum
GT-MHR – Gas Turbine Modular Helium-cooled Reactor
HTR – High Temperature Reactor
HTR-10 – High Temperature Test Reactor (10 MW_e)
HTR-PM – High Temperature Reactor – Prototype Modular
HTR-TN – High Temperature Reactor Technology Network
HTTR – High Temperature Test engineering Reactor
IHX – Intermediate Heat eXchanger
INET – Institute of Nuclear and New Energy Technology
INL – Idaho National Laboratory
JAEA – Japan Atomic Energy Agency
KAERI – Korea Atomic Energy Research Institute
MHTGR – Modular High Temperature Gas-cooled Reactor
MW_e – Megawatt (electric)
MW_{th} – Megawatt (thermal)
NGNP – Next Generation Nuclear Project
NHDD – Nuclear Hydrogen Development and Demonstration
NRC – Nuclear Regulatory Commission
PBMR – Pebble Bed Modular Reactor
PIRT – Phenomena Identification and Ranking Table
R&D – Research and Development
SNE-TP – European Sustainable Nuclear Energy Technology Platform
THTR – Thorium High Temperature Reactor
TRISO – Tri-Structural Isotropic Fuel
US-DOE – Department Of Energy of the United-States
VHTR – Very High Temperature Reactor

References

1. ZHANG, Z., Z. Wu *et al.*, "Current Status and Technical Description of Chinese 22 x 250 MW_{th} HTR-PM Demonstration Plant", "Nuclear Engineering and Design", Vol. 239, Issue 7, July 2009.

2. KRIEK, J., "Status of the Pebble Bed Modular reactor Project", *Proc. of the 4th International Topical Meeting on High Temperature Reactor Technology HTR2008,* Washington, DC (USA) (2008).

3. HOLBROOK, M.R. and T. Cook, "NRC Licensing Strategy Development for the NGNP", *Proc. of the 4th International Topical Meeting on High Temperature Reactor Technology HTR2008,* Washington, DC (USA) (2008).

4. KUNITOMI, K., *et al.*, "JAEA's VHTR for Hydrogen and Electricity Cogeneration: GTHTR300C", "Nuclear Engineering and Technology", Vol. 39 No. 1 February 2007.

5. CHANG, J., "Status of Nuclear Hydrogen Production Technology Development Project in Korea", International Conference on Non-electrical Applications of Nuclear Power, Oarai (Japan) (2007).

6. Sustainable Nuclear Energy Technology Platform – Strategic Research Agenda 2009 (www.SNETP.eu).

7. Generation IV International Forum Annual Report (2008), OECD Nuclear Energy Agency, Issy-les-Moulineaux (2008).

8. Malo, J.Y., *et al*., "Gas-cooled Fast Reactor 2 400 MW_{th}, end of the preliminary viability phase", ICAPP'08, Anaheim, USA, June 8-12, 2008.

9. Bertrand, F., *et al*., "Preliminary Safety Analysis of the 2 400 MW_{th} Gas-Cooled Fast Reactor", ICAPP'08, Anaheim, USA, June 8-12, 2008.

10. Dumaz, P., *et al*., "Studies of unprotected transients and alternative decay heat removal system for the gas-cooled fast reactor", ICAPP'08, Anaheim, USA, June 8-12, 2008.

ADVANCED SUPERCRITICAL WATER AND MOLTEN SALT REACTORS

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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
VHTR SA	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
SFR SA		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
SCWR SA	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						
GFR SA		X	X	X					X	
GFR FCMFC Project		P	P	P					O	
GFR CD & S Project		P	P						P	
LFR System		P		P						P
MSR System		P	P							P

X = Signatory P = Provisional participant O = Observer

Acronyms of Projects

HP	Hydrogen Production
FFC	Fuel and Fuel Cycle
CMVB	Computational Methods Validation and Benchmarking
MAT	Materials
AF	Advanced Fuel
GACID	Global Actinide Cycle International Demonstration
CDBOP	Component Design and Balance-Of-Plant
SO	Safety and Operation
SIA	System Integration and Assessment
M&C	Materials and Chemistry
TH & S	Thermal-Hydraulics and Safety
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 4th 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. 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.

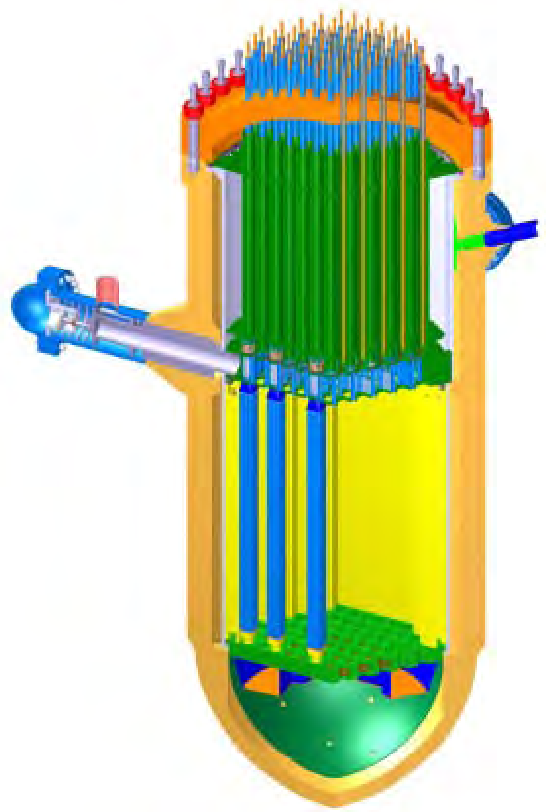

Figure 1. Design of the High Performance Light Water Reactor, Schulenberg and Starflinger [1]

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_2, 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.

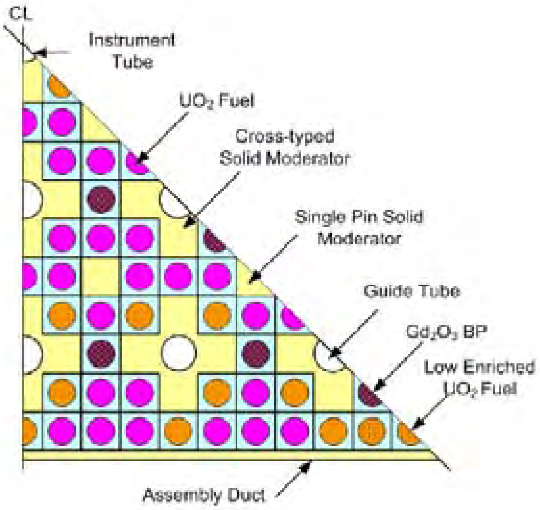

Figure 2. Fuel assembly design of a 1 400 MWe core, Bae *et al*. [4]

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 experi-mental 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.

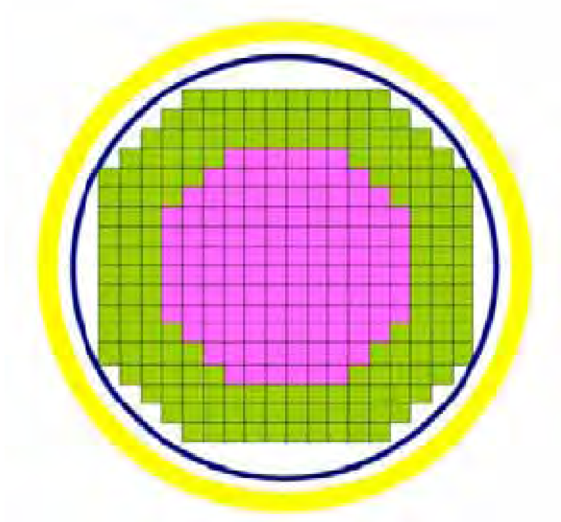

Figure 3. Scheme of the mixed SCWR core, Cheng [5]

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_2). 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]

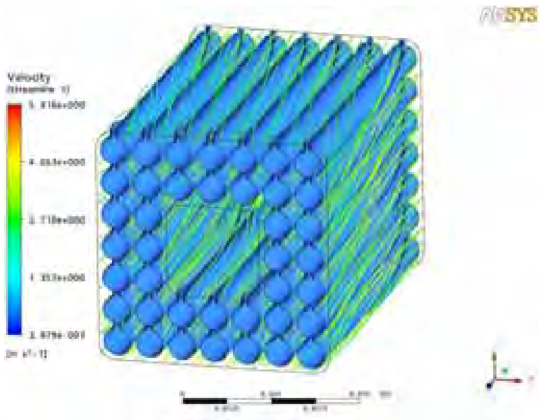

Figure 4. Streamlines of flow through a SCWR fuel assembly with wire wrapped fuel rods, Kiss *et al*. [6]

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 6th Framework Programme. A continuation study is proposed as a contribution to the 7th 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", 4th 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", 4th 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", 4th 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", 4th Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
5. Cheng, X., "R&D activities on SCWR in China", 4th 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", 4th 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 7th 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.

“TOWARDS INDUSTRIAL IMPLEMENTATION: PUBLIC AND PRIVATE INITIATIVES INTERCONNECTIONS”

REMARKS

Edward McGinnis *(Panel Chair)*

U.S. Department of Energy (*Edward.McGinnis@hq.doe.gov*)

Panelists in the industry session of the GIF Symposium discussed both the private-sector and government perspectives and roles for the commercial deployment of Generation IV International Forum reactor concepts.

The dialogue focused on specific areas that should be taken into consideration as the nuclear renaissance moves beyond Generation III reactors to deployment of Generation IV reactors.

The specific areas covered during the private-sector discussions were safety and market requirements; innovative and collaborative developments, sustainability and economics.

As the nuclear industry implements deployment to commercialization, Generation IV reactor’s safety case must be clear, transparent and convincing to nuclear regulators in countries exploring the need for these reactors and the general public.

Industry and government must partner in the development process to ensure innovative and collaborative technical advances while minimizing technical, market, and financial risks.

Leveraging industry collaboration with government can move technology more quickly to a stage of maturity that provides more acceptable risk profiles for financing.

As they emerge into the market, Generation IV reactors should remain economically comparable not only to other energy sources but to modern Generation III + reactors.

As Generation IV reactors come to the market, there will be significant shifts in key sectors of the reactor market globally, increasing costs and overall financing and increasing demand for human capital infrastructure to support construction, operation, maintenance and inspection/regulation of the new reactors.

The specific areas covered during the government discussion were Japan’s Sodium Fast Reactor approach, the French strategy and the United States approach.

Japan’s Sodium Fast Reactor approach includes early establishment of roles and duties between the public and private sector during the development project: integration of utilities and vendors at the early stages, to include at the conceptual design phase, to further enhance safety, reliability, sustainability, non-proliferation and economic competitiveness.

The French presentation of their strategy, described in multiple phases, outlines their approach to secure the current fleet of power water reactors: deployment of 4th generation fast reactors and waste management, and participation in ITER for fusion technology.

The U.S. approach described the Department of Energy's focus on trans–formational research in the areas of nuclear science and technology to address climate change and energy security. One highlight is the expansion of the U.S. Generation IV research and development to solve underlying technology challenges of advanced reactor concepts. The expanded areas of research and development are design development, support for new regulatory framework and government and industry partnership on design development.

OPENING SESSION

Chair: Massimo Salvatores

An Overview of Generation IV Strategy and Outlook

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Paper presented by H. McFarlane

I. INTRODUCTION

The importance of reducing greenhouse gas emissions is now almost universally recognized by national policies. Numerous strategies and scenarios are proposed in order to achieve more sustainable future energy supplies. In the majority of these scenarios, nuclear's growth is an essential element. For example the 2008 World Energy Outlook forecasts an additional 250 GWe of nuclear capacity by 2030[1] in a scenario that would stabilize the atmosphere at 450 ppm CO_2 and thereby limit global warming to 2°C above pre-industrial levels. In such a scenario baseload nuclear would complement other forms of clean energy, which are subject to variability, intermittency, and low power density.

Many nations, both heavily industrialized and emerging economies, are driving the growth of nuclear energy. Some 43 new units are under construction in 11 countries, with more projects preparing to move forward.[2] Nevertheless, challenges still exist to further large-scale deployment of nuclear: (1) nuclear energy must become more sustainable from the standpoint of its utilization of nuclear fuel resources as well as the management and disposal of nuclear waste, (2) the units must operate reliably and be economically competitive, (3) their safety must remain of paramount importance, and (4) nuclear deployment must be undertaken in a manner that does not add to concern about proliferation of nuclear weapons. In addition, new technologies should help meet anticipated future needs for a broader range of energy products beyond electricity, and governments should support the revitalization of their nuclear R&D infra–structure.

To meet these challenges and deliver future nuclear energy systems, the Generation IV International Forum is undertaking some of the R&D necessary to develop the next generation of innovative nuclear energy systems that can supplement today's nuclear plants and transition nuclear energy into the long term. Generation IV nuclear energy systems comprise the nuclear reactor and its energy conversion systems, as well as the necessary facilities for the entire fuel cycle from ore extraction to final waste disposal.

I.A. Strategy

The Forum's strategy has been to (1) define challenging goals for next generation systems and identify viable candidate technologies that may address them by about 2030, (2) gain participation of the countries leading the world's nuclear development and create a legal framework for their multilateral cooperation, and (3) organize and grow the program and further stimulate the world research community to join the effort.

The first part was addressed in 2000-2002, culminating in the Generation IV Roadmap,[3] that evaluated many concepts and recommended six systems. The second was addressed in 2003-2005, culminating in the Framework Agreement,[4] a legally binding instrument of the Members that provides for cooperative exchange, creation, ownership and protection of intellectual property in multilateral research contracts. The third is presently taken up with a variety of communications and interactions. The period of 2006-2009 has seen considerable R&D planning and organization, the results of which are being the Forum's Policy Group has recently conducted an exercise in strategic planning that revisits all areas with the expectation of identifying needed changes and actions that keep pace with the changing world situation. This paper reports their results.

I.B. Outlook

In brief, it has been a little more than seven years since the Roadmap was published, and four years since the first signing of the Framework. The former heralded what to work on, the latter provided for how, and the Forum now addresses the question of when by describing the expected accomplishments and emphasis of the next five years in the outlook for the future.[5]

II. SYSTEM TECHNOLOGIES

Each Forum member is free to choose the systems that they will advance, as well as to pursue any options or alternatives to the systems outside of the formally agreed System Research Plan. To understand the various organizational entities that are mentioned here (steering committees, project management boards, methodology working groups, etc.), an overview can be found on the Generation IV website.[6]

With respect to the six Generation IV systems, presented in order of their level of cooperative activity within the Framework today, the Forum expects the following progress in five years.

II.A. VHTR

For the very high temperature gas-cooled reactor (VHTR), the full complement of technology projects will have been created. Feasibility issues regarding hydrogen production, fuel performance, and high temperature design including both the core and intermediate heat exchanger will be resolved, or nearly so. An assessment of progress toward the goals will have been completed for the major options. Key performance issue tests will be in planning, with some in operation, and decisions will have been made about advancing one or more prototypes.

II.B. SFR

For the sodium-cooled fast reactor (SFR), the full complement of technology projects will also have been created. Feasibility issues regarding full actinide recycling with multiple passes, competitive capital cost, in-service inspection and repair, and alternate energy conversion (*e.g.*, with gas or supercritical CO_2 Brayton cycles) will be resolved, or nearly so. An assessment of progress toward the goals will have been completed for the major options. Key performance issue tests will be in planning, with some in operation, and decisions will have been made about advancing one or more prototypes. The Russian SFRs BOR-60 and BN-600 continue to provide long-term operating data. Fresh operating experience is anticipated to be gathered from new SFRs in various countries and from the restart of MONJU.

II.C. SCWR

For the supercritical water-cooled reactor (SCWR), a set of essential technology projects will have been created. Feasibility issues regarding core layout and spectrum, fuel forms and possible recycling, in-core materials behavior, and system thermal-hydraulics and safety will be much better understood and on their way to resolution. The SCWR will be nearing a point at which it may assess its progress toward the goals. Key viability tests will be in operation.

II.D. GFR

For the gas-cooled fast reactor (GFR), a set of essential technology projects will also have been created. Feasibility issues regarding fuel forms and actinide recycling, system safety and analysis, and cost will be much better understood and on their way to resolution. The GFR will be nearing a point at which it may assess its progress toward the goals. Key viability tests will be in operation.

II.E. LFR

For the lead-cooled fast reactor (LFR), formal collaborations will have begun, and a set of exploratory projects will have been created. Feasibility issues regarding coolant and materials, energy conversion and components, actinide recycling and system safety will be much better understood and preparations for viability testing will be underway.

In Europe it is expected that a choice between gas or a heavy liquid metal coolant for fast reactors, as a possible alternative to sodium, will be made with the potential launch of an experimental reactor using the selected coolant.

II.F. MSR

For the molten salt reactor (MSR), formal collaborations will also have begun, and a set of exploratory projects will have been created. Feasibility issues regarding its fuel cycle, salt chemistry with dissolved fuel isotopes (including transuranics) and materials compatibility will be much better understood and preparations for viability testing will be underway. Issues on the operation and safety of the coupled MSR reactor and fuel processing unit will be clarified.

II.G. Crosscutting R&D

R&D synergies will be developed between system steering committees, in domains such as requirements, design rules and codes, equipment, instrumentation, components and subsystems.

Generation IV is focused on four performance goals, related to safety and reliability, proliferation resistance and physical protection, economics, and sustainability. Three crosscutting methodology working groups have been created to develop evaluation methods that can assess the performance of new designs toward the Generation IV goals. During the coming five years these working groups will continue to support the six system steering committees in evaluating and guiding the optimization of their system designs. In addition, support for revitalizing and developing nuclear R&D infrastructure in terms of facilities, people and new advanced simulation and validation tools will be emphasized.

III. MISSIONS AND RESOURCES

The Forum is monitoring the scope and pacing of its research portfolio to keep in tune with global developments. As a result, several missions for the systems are expected to be given increased emphasis or otherwise modified to reflect future trends.

III.A. Hydrogen and Process Heat

While there is much debate about when or even if a large-scale deployment of a hydrogen economy may happen, it is now well understood how vital a role hydrogen currently plays in the production of premium transportation fossil fuels and chemical feedstocks. At the same time, there is a growing interest in the utilization of high-temperature systems to high-temperature process applications. The Forum has encouraged its high-temperature systems to broaden their mission to include process heat applications more generally. This is an important way to make nuclear energy more relevant as a non-greenhouse gas emitting source of primary energy beyond electricity.

III.B. Water Desalination

Second, in recent years there is a growing awareness of water shortages in many regions of the world. While the missions of Generation IV have included electricity, hydrogen production and actinide management in the original Roadmap, we may be nearing a time at which desalination should be highlighted in the missions if current generation reactors cannot successfully address it. The Forum will continue to monitor this, as the development of such new energy products that can expand nuclear energy’s benefits beyond electrical generation contribute to the sustainability goals of Generation IV.

III.C. Small Reactors

Third, there is a growing interest in addressing the needs of countries that are better served by smaller systems. While a few options within Generation IV systems are being pursued with small module size, they are intended to complement the evolutionary designs of industry for near-term deployment, and thereby provide for the long term future need. Of course, the specific technologies developed in Generation IV (such as new materials, fuels or energy conversion technology) may beneficially diffuse into the evolutionary designs in advance of a next generation.

III.D. Fuel Resources

Fourth, from the perspective of uranium resource conservation, many of the Generation IV systems investigated are fast neutron reactors which use plutonium and uranium recovered from spent fuel by reprocessing, and depleted uranium. However, the Generation IV steering committees have shown increasing interest in the use of thorium resources. In fact, we are already seeing some exploration of thorium-based fuels in some Generation IV systems to understand their potential benefits. The Forum encourages measured pursuit of this alternative by systems to the extent that it allows them to advance toward the sustainability goal.

IV. TECHNICAL COOPERATION AND MEMBERSHIP

Technical cooperation and engagement of the research community worldwide plays a key role in the successful development of Generation IV systems. In the next five years, the Forum will expand the number of topical sessions that are sponsored. These will bring news of technical interests, research problems and breakthroughs to the research community with the intent of stimulating more participation by academia, industry and laboratories. Second, the Forum will monitor the level of funded collaborations by industry, and increase it significantly. Third, the Forum will continue to harmonize the efforts of its members on major technology demonstrations, such as is being done with several sodium reactor demonstration projects today.

Finally, note that the Forum's membership has changed over the years. While among the original signatories to the Generation IV Charter, Argentina and Brazil have made the decision to become inactive in the Forum largely as a matter of their research priorities. The United Kingdom also decided to become inactive, although the government still allows their technical community to participate in Generation IV through EURATOM. More recently, in 2006, China and Russia are the newest signatories to the Charter. In regards to the Framework Agreement, China acceded in 2007, the Republic of South Africa acceded in 2008, and Russia plans to accede in 2009. The original intent of the Forum remains the same to bring the collaborative efforts of the major developers of next generation nuclear energy systems to bear in a concerted effort. The Forum welcomes the prospect of additional members that can bring significant resources and capabilities, and hopes to report the successful entry of a few new members over the next five years.

V. CONCLUSION

The Generation IV International Forum's resolve is to deliver future nuclear energy systems that enable the safe, sustainable worldwide growth of nuclear energy well into the future for the benefit of mankind. Optimistic about the long term prospects for nuclear energy, the Forum plans to contribute to its success.

Acknowledgements

The author gratefully acknowledges the participation of the Policy Group representatives, and especially the Chair, Jacques Bouchard, in the vision they have set forth.

References

1. International Energy Agency, *World Energy Outlook: 2008*, Organisation for Economic Cooperation and Development, Paris (2008).
2. World Nuclear Association, *World Nuclear Reactors*, www.world-nuclear.org/info/reactors.html, project data updated 5 Feb (2009).
3. Generation IV International Forum, *A Technology Roadmap for Generation IV Nuclear Energy Systems*, GIF-002-00, Idaho National Laboratory (2002).
4. Generation IV International Forum, *Framework Agreement*, www.gen-4.org/PDFs/Framework-agreement.pdf, signed 28 Feb (2005).
5. Generation IV International Forum, *GIF R&D Outlook*, in preparation (2009).
6. www.gen-4.org.

SESSION I

METHODOLOGY OVERVIEWS AND FOCUS ON APPLICATIONS

VERY HIGH TEMPERATURE REACTOR (VHTR)

Co-Chairs: Moon H. Chang and Gideon Greyvenstein

COST ESTIMATING METHODOLOGY AND APPLICATION

W.H. Rasin[(1)] and K. Ono[(2)]

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I. INTRODUCTION

The Economic Modeling Working Group (EMWG) was created by Generation IV International Forum (GIF) early in 2003. The Group was charged with developing a methodology to assess the progress of the Generation IV systems in achieving the economic goals established by the GIF Policy Group. The objective was to establish a simplified cost estimating methodology appropriate for Generation IV systems in various stages of development and sufficiently rigorous to promote consistent application by the systems development groups. The EMWG is working with the System Steering Committees to provide training and assistance in the application of the methodology.

The GIF Cost Estimating Methodology has been developed and tested by the EMWG. It has been released for general use by the GIF System Steering Committees. The Policy Group, at the request of the EMWG, agreed to release the methodology to the general public to achieve more widespread experience with its application.

The Cost Estimating Methodology consists of (1) the Generation IV Cost Estimating Guide-lines and (2) a software package, G4-ECONS, to facilitate the implementation of the Guidelines.

The EMWG monitors the application of the methodology and continues to assess economic trends and experience which may have economic impacts on Generation IV systems.

II. GENERATION IV ECONOMIC GOALS

Early in the Generation IV process, the GIF Policy Group established a comprehensive set of goals to guide the development of Generation IV systems. Among the goals are two economic goals:

- to have a life cycle cost advantage over other energy sources (*i.e.*, to have a lower levelized unit cost of energy on average over their lifetime);
- to have a level of financial risk comparable to other energy projects (*i.e.*, to involve similar total capital investment costs and capital at risk).

III. GIF COST ESTIMATING GUIDELINES

The GIF Cost Estimating Guidelines provide a comprehensive approach for assessing the performance of Generation IV systems in achieving the established economic goals. The methodology may be used to assess if the Generation IV systems are indeed improved over Generation III or to improve the cost of Generation IV systems on a sub system level as the development proceeds. The Guidelines proceeded through several revisions as the methodology was developed and tested. Revision 4 is the current version and was released to GIF and the public in 2007.

The Guidelines provide detailed processes for developing the total capital investment cost and calculating the levelized unit electric cost. The overall structure of the cost estimating methodology is shown in Figure 1.

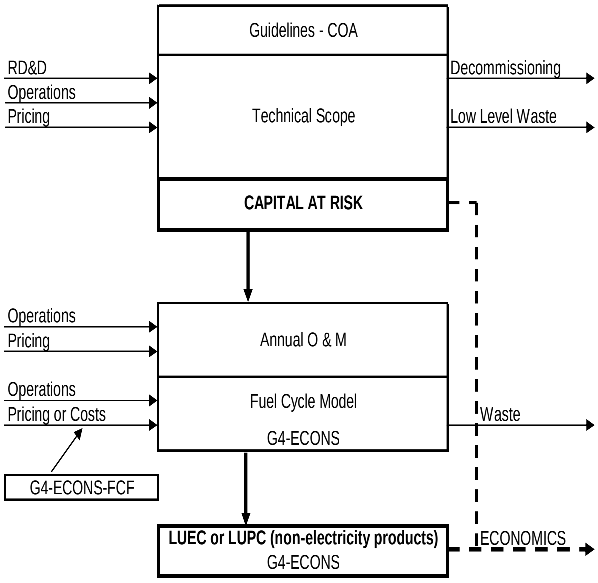

Figure 1: Structure of the GIF Cost Estimating Methodology

The central feature of the methodology is the comprehensive Code of Accounts. The Code of Accounts provides a disciplined structure for capturing and categorizing all appropriate costs in the development of consistent system cost estimates. An overview of the Code of Accounts is given in Chapter 1 of the Guidelines and a sample of a detailed Code of Accounts "dictionary" is provided in Appendix F.

Chapter 3 of the Guidelines provides a Code of Accounts for Research, Development and Demonstration costs that precede the actual system design. Such costs are usually not included in a system cost estimate but are important considerations for management in assessing the overall development cost for a given system. To date, this RD&D Code of Accounts has not been employed by any of the GIF system development groups.

Because the Generation IV systems will for some time be in varied states of development and maturity, two different approaches for cost estimation are included. Chapter 4 of the Guidelines describes a "bottom up" approach appropriate for systems in an advanced state of development with some degree of design detail. The "bottom up" approach yields the most reliable and complete cost estimate and should be the ultimate outcome for a cost estimate on a mature system. Since most Generation IV systems will for some time be in a less mature state of development, Chapter 5 describes a "top down" method of cost estimation appropriate for use with evolving system development.

IV. G4-ECONS

To facilitate implementation of the Cost Estimating Guidelines, the EMWG developed an EXCEL based spreadsheet package, G4-ECONS. G4-ECONS 2.0 was released to GIF and the public in 2008. The software package facilitates the input of total capital cost at a high level to prevent the inadvertent disclosure of proprietary data. Levelized unit electric cost is also calculated. G4-ECONS 2.0 provides the capability for cost estimates of systems designed for other than electricity production, such as desalination or hydrogen production. Companion software, G4-ECONS-FCF, provides the capability to calculate cost of product from any fuel cycle facility. The basic structure of G4-ECONS is shown in Figure 2.

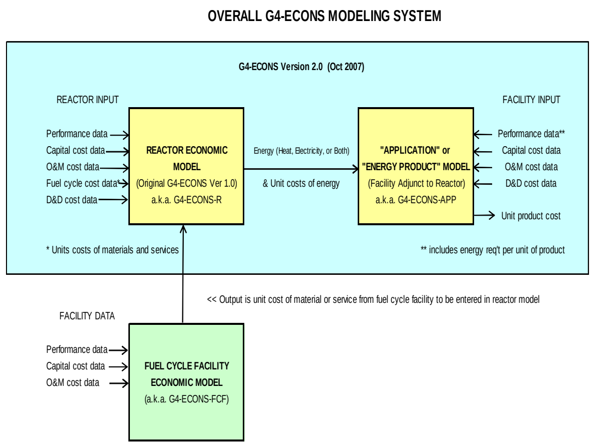

Figure 2: Structure of G4-ECONS

V. APPLICATIONS OF THE GIF COST ESTIMATING METHODOLOGY

During the development phase of the methodology, the EMWG performed analysis of a Generation III CE System 80+ reactor to compare the results to published cost evaluations. The methodology, particularly the G4-ECONS software, compared well with published results. The levelized cost of electricity calculated by G4-ECONS was within 1% of the published figure. This test served as an initial validation of the software.

The first Generation IV trial application was performed by the Japanese members of the EMWG for the Japanese Sodium Fast Reactor (JFSR). These results were presented at the ANS meeting in Boston in June of 2007. Again the cost estimate comparisons were quite good thus providing a validation for Generation IV application. The G4-ECONS output for the JSFR is shown in Table 1.

case: JSFR Sample Calculation/ April 14,2006 (Closed cycle)		
Worksheet name: LUEC SUMMARY		
TOTAL REACTOR & FUEL CYCLE SYSTEM		
Summary of Model Results		
Case: JSFR Sample Calculation/ April 14,2006 (Closed cycle)		
Discount rate=	2.00%	
	Annualized Cost in $M/yr	Mills/kwh or $/MWh
Capital Cost incl Financing	$77.4	6.51
Operations Cost	$88.6	7.46
Fuel Cycle Cost	$77.8	6.55
D&D Cost	$5.1	0.43
Totals	$249.0	**20.95**

Table 1: G4-ECONS Output Screen for JSFR Sample Calculation

The GIF System Steering Committees are preparing to apply the Cost Estimating Methodology to their projects as the designs become sufficiently complete to do so. The methodology is also being applied by other groups at the International Atomic Energy Agency and several Universities for both existing and advanced designs. Results of these applications are beginning to be published at various technical meetings and conferences.

The cost comparison with the Japanese cost estimating codes is shown in Table 2.

Mills/kWh or $/MWh	G4-ECONS Code	FCC-EX*1	Final Report - FS Phase-2
Estimated year (year)	2005	2005	2005
Capital Cost incl. Financing	6.08	5.94	6.44
D&D Cost	0.43	0.43	0.43
Operations Cost	7.46	7.33	7.31
Front-end Fuel cycle	1.46	1.46	1.50
Back-end Fuel Cycle	5.09	5.09	2.99
Initial core fuel front-end Cost*2	0.43	0.23	0.41
Initial core back-end Cost*2	-	0.81	0.88
Totals	20.95	21.30	19.96

Table 2: G4-ECONS Comparison with Japanese Codes

The GIF Cost Estimating Methodology is available on a compact disk which includes the GIF Cost Estimating Guidelines, G4-ECONS Users Manual, G4-ECONS and G4-ECONS-FCF software. The disk may be obtained by emailing the Organization of Economic Cooperation and Development: webmaster@g4if.org.

The EMWG is tracking the distribution and monitoring the application of the methodology. Further improvements and revisions will be undertaken as experience indicates that such changes would be advantageous for specific GIF applications.

VI. CONCLUSION

The Generation IV Cost Estimating Methodology was developed to promote consistent evaluations of Generation IV Systems with respect to the economic goals set by the Policy Group. It has been tested against both Generation III and Generation IV systems and demonstrated to be a valid methodology for system cost estimation. The methodology is available to both GIF and non-GIF users. The EMWG will continue to track the application of the methodology and make improvements as the needs of GIF may indicate. Training and assistance in application of the methodology is provided as requested by the System Steering Committees.

Acknowledgements

This paper is presented on behalf of the GIF EMWG and represents the work done by its members.

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References

1. Generation IV Cost Estimating Guidelines Revision 4, GIF EMWG, 2007. Available from OECD NEA.
2. G4ECONS G4-ECONS Version 2.0, GIF EMWG, 2008. Available from OECD NEA.
3. G4ECONS G4-ECONS Users Manual, GIF EMWG, 2008 Available from OECD NEA.
4. JAEA Sodium Cooled Fast Reactor (JSFR) Total System Cost Analysis using G4-ECONS, Kiyoshi Ono, ANS Meeting, Boston, 2007

PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION EVALUATION METHODOLOGY DEVELOPMENT AND APPLICATIONS

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Abstract

We present an overview of the technical progress and accomplishments on the evaluation methodology for proliferation resistance and physical protection (PR&PP) of Generation IV nuclear energy systems. We intend the results of the evaluations performed with the methodology for three types of users: system designers, program policy makers, and external stakeholders. The PR&PP Working Group developed the methodology through a series of demonstration and case studies. Over the past few years various national and international groups have applied the methodology to nuclear energy system design as well as to developing approaches to advanced safeguards.

I. INTRODUCTION

We present the technical progress and accomplishments on the evaluation methodology for proliferation resistance and physical protection (PR&PP) of advanced nuclear energy systems (NESs). The Generation IV Roadmap [1] recommended the development of an evaluation methodology to define measures for PR&PP and to develop a methodology for evaluating them for the six NESs proposed within the Generation IV program. Accordingly, the Generation IV International Forum (GIF) formed a Working Group in December 2002 to develop a methodology. GIF approved the current version of the methodology (Revision 5) for open distribution and it is available at the GIF website. [2]

For a proposed NES design, the methodology defines a set of challenges, analyzes system response to these challenges, and assesses outcomes. The challenges to the NES are the threats posed by potential actors (proliferant States or sub-national adversaries). The characteristics of Generation IV systems, both technical and institutional, are used to evaluate the response of the system and to determine its resistance against proliferation threats and robustness against sabotage and terrorism threats. The outcomes of the system response are expressed in terms of a set of measures, which are the high-level PR&PP characteristics of the NES. The methodology is organized to allow evaluations to be performed at the earliest stages of system design and to become more detailed and more representative as the design progresses. It can thus be used to enable a program in safeguards by design or to enhance the conceptual design process of an NES with regard to intrinsic features for PR&PP. We intend the results of the evaluations performed with the methodology for three types of users: system designers, program policy makers, and external stakeholders.

II. OBJECTIVES AND OVERVIEW OF ASSESSMENT APPROACH

The Technology Goals for Generation IV nuclear energy systems (NESs) highlight Proliferation Resistance and Physical Protection (PR&PP) as one of the four goal areas along with Sustainability, Safety and Reliability, and Economics:

Generation 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.

We define PR&PP as follows:

Proliferation resistance is that characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices.

Physical protection (robustness) is that characteristic of an NES that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices (RDDs) and the sabotage of facilities and transportation by sub-national entities and other non-Host State adversaries.

According to the current Terms of Reference approved by GIF, the responsibilities of the PR&PP Working Group (WG) are as follows:

- Maintain cognizance of PR&PP evaluations conducted under the auspices of GIF or with the knowledge and counsel of GIF through its member states, and serve as a clearinghouse for advice to the GIF Policy and Experts Groups on PR&PP issues related to Generation IV nuclear energy systems;
- Monitor the integrity and quality of evaluations conducted under the auspices of GIF or with the knowledge and counsel of GIF through its member states under terms and conditions that protect proliferation-sensitive and proprietary information, provide peer review of PR&PP evaluations upon request, and address questions related to the fidelity with which the methodology is applied;
- Maintain configuration control over the PR&PP methodology, its documentation and revisions, and serve as a central authority to review and accept methodology improvements and incorporate them in the configuration controlled GIF PR&PP methodology;
- Strengthen the link with Generation IV system designers, in particular with GIF System Steering Committees;
- Maintain cognizance of and interactions with other GIF related activities, such as the Risk and Safety Working Group;
- Maintain cognizance of and interactions with non-GIF activities such as IAEA initiatives and specific national initiatives;
- Promote and facilitate early consideration of PR&PP in the development and design of Generation IV systems;
- Promote PR&PP goals and broad acceptance of the PR&PP methodology by participation in conferences and publication of papers;
- Maintain capability to perform or direct PR&PP studies on request of GIF.

The diagram shown here illustrates the methodological approach at its most basic. As noted in the Introduction, for a given system, analysts define a set of ***challenges***, analyze ***system***

CHALLENGES → **SYSTEM RESPONSE** → **OUTCOMES**

Threats ***PR & PP*** ***Assessment***

response to these challenges, and assess ***outcomes***.

The evaluation methodology assumes that an NES has been at least conceptualized or designed, including both the intrinsic and extrinsic protective features of the system. Intrinsic features include the physical and engineering aspects of the system; extrinsic features include institutional aspects such as safeguards and external barriers. A major thrust of the PR&PP evaluation is to elucidate the interactions between the intrinsic and the extrinsic features, study their interplay, and then guide the path toward an optimized design.

The structure for the PR&PP evaluation can be applied to the entire fuel cycle or to portions of an NES. The methodology is organized as a *progressive* approach to allow evaluations to become more detailed and more representative as system design progresses. PR&PP evaluations should be performed at the earliest stages of design when flow diagrams are first developed in order to systematically integrate proliferation resistance and physical protection into the designs of Generation IV NESs along with the other high-level technology goals of sustainability, safety and reliability, and economics. This approach provides early, useful feedback to designers, program policy makers, and external stakeholders from basic process selection (*e.g.,* recycling process and type of fuel), to detailed layout of equipment and structures, to facility demonstration testing.

III. RECENT ACCOMPLISHMENTS

The PR&PP WG has recently performed a case study on an example sodium fast reactor (ESFR) and its associated fuel cycle to exercise the methodology and to obtain preliminary insights on the PR&PP aspects of this system [3]. There is also an ongoing effort [4] to seek harmonization between the PR&PP methodology and an initiative by the International Atomic Energy Agency on a related approach to proliferation resistance that has been developed under the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). The purpose of this harmonization activity is to more fully understand and articulate the range of applicability and the potential for appropriate synergy and cooperation among the two efforts. Further, the PR&PP WG and the System Steering Committees (SSCs) for each of the six design concepts within GIF have undertaken a focused effort to integrate PR&PP notions into the design activities for each of the six concepts.

Example Sodium Fast Reactor Case Study

The PR&PP WG has developed its methodology with the aid of a series of studies. The ESFR consists of four sodium-cooled fast reactors of medium size co-located with an on-site dry fuel storage facility and a pyrochemical spent fuel reprocessing facility.

The objectives of the Case Study were to exercise the GIF PR&PP methodology for a complete Generation IV reactor/fuel cycle system; to demonstrate, via the comparison of different design options, that the methodology can generate meaningful results for designers and decision makers; to provide examples of PR&PP evaluations for future users; to facilitate transition to other studies; and to facilitate other ongoing collaborative efforts (*e.g.,* INPRO) and other national efforts.

Lessons learned were that each PR&PP evaluation should start with a qualitative analysis allowing scoping of the study, of the assumed threats and identification of targets, system elements, etc.; that there is a need to include detailed guidance for qualitative analyses in methodology; that the role of experts is essential; that there is a need for PR and PP experts and expert elicitation techniques; and that qualitative analysis offers valuable results, even at the preliminary design level. Qualitative analysis can directly address the measures for PR: Technical Difficulty (TD), Proliferation Time (PT), Proliferation Cost (PC), and Material Type (MT). However, Detection Resource Efficiency (DE) and especially Detection Probability (DP) are harder to quantify using qualitative analysis.

Systematic identification of potential diversion pathways is a key goal. We found that it is possible to systematically identify targets and potential pathways for each specific threat, and to systematically search for plausible scenarios that could implement the potential proliferant Host

State's strategies to divert the target material. A set of diversion pathway segments were developed and the proliferation resistance measures for each pathway were determined. The methodology compares and distinguishes how different design choices affect proliferation resistance.

The diversion pathways analysis provides a variety of useful information to stakeholders, including regulatory authorities, government officials, and system designers. This information includes how attractive the material is to potential proliferators for use in a weapons program; how difficult it would be to physically access and remove the material; and whether the facility can be designed and operated in such a manner that all plausible acquisition paths are impeded by a combination of intrinsic features and extrinsic measures.

The misuse pathways analysis requires consideration of potentially complex combinations of processes to produce weapons-usable material, *i.e.*, it is not a single action on a single piece of equipment, but rather an integrated exploitation of various assets and system elements. We found that, given a proliferation strategy, some measures are likely to dominate over the others, and within a measure some segments will dominate the overall pathway estimate.

The breakout pathways analysis found that breakout is a *modifying strategy* within the diversion and misuse threats and can take various forms that depend upon intent and aggressiveness, and ultimately the proliferation time assumed by a proliferant state. Furthermore, measures can be assessed differently within the breakout threat, depending upon the breakout strategy chosen. Some additional factors related to global response and foreign policy were identified as being relevant to the breakout threat, but those factors are not included in the PR&PP methodology.

The theft and sabotage pathways analysis found that multiple target and pathways exist. The most attractive theft target materials appeared to be located in a few target areas. Specifically, for the ESFR, the most attractive theft target areas with the most attractive target materials were found to be the LWR spent fuel cask parking area, the LWR spent fuel storage and fuel cycle facility staging/washing area, the fuel cycle facility air cell (hot cell), and the inert hot cell.

As noted in the PR&PP methodology report, [2] a substantial base of analytic tools already exists for theft and sabotage pathway analysis. The case study verified that these tools can be used within the paradigm of the PR&PP methodology.

The Case Study indicated that the methodology could be improved by:

- Applying the measures to a broader range of targets and pathways to gain additional experience with their practical application,
- Investigating the specific form of the metrics used to express the measures.

Interactions with Nuclear Energy System Designers

As part of the effort to familiarize GIF participants with the PR&PP methodology, particularly system designers and program policy makers and to better understand the needs of the designers, a series of workshops were held beginning in the US in 2005, Italy in 2006, Japan in 2006, and Republic of Korea in 2008. Useful mutual information exchange occurred during these workshops which helped to further define the methodological approach and the needs of the users.

Also, in 2007 informal discussions began between the PR&PP WG and representatives of the GIF System Steering Committees (SSCs) for each of the six Generation IV design concepts on the exploration of ways that the two entities could cooperate in the assessment and enhancement of PR&PP performance of Generation IV systems. A workshop of interested parties was held in May 2008 at Brookhaven National Laboratory which resulted in a program plan for future joint activities. Three broad goals were defined for future joint activities: 1) identify in the near term salient features of the design concepts that impact their PR&PP performance, 2) perform cross-cutting studies that assess against PR&PP criteria design or operating features common to various

Generation IV systems, and 3) infer functional requirements for the global layout of future nuclear energy systems. See paper by F. Carré and S. Felix, Proceedings of Global 2009, for further details. [5]

As of this writing, draft white papers on the PR&PP aspects and issues of each of the six design concepts are in development between representatives of the SSCs and the PR&PP WG. A follow-on workshop is planned for July 2009 to further advance the white papers and to continue future joint activities.

Interactions with GIF RSWG

In addition to the establishment of the PR&PP WG, the GIF has recognized the need for a Risk and Safety Working Group (RSWG) to address the approach to be adapted to safety of future nuclear energy systems. The GIF also recognized that an interface with the activities of the PR&PP WG would be needed, and thus noted:

- A need for integrated consideration of safety, reliability, proliferation resistance and physical protection approaches in order to optimize their effects and minimize potential conflicts between approaches.
- A need for mutual understanding of safety priorities and their implementation in PR&PP and RSWG evaluation methodologies.

The efforts of these two groups continue to be carefully coordinated. This has been largely accomplished so far via the close working relations between the leaderships of the two groups. Advances by either group have relevance to the other and are mutually beneficial to both. It also continues to be important to assess and understand the impact of all specific design features in relation to objectives of safety performance, physical protection, and proliferation resistance.

See Khalil *et al*., Proceedings of Global 2009 for further details. [6]

Proliferation Risk Reduction Assessments

Assessments of proliferation risk reduction are being conducted in various countries that participate in GIF as part of their respective national programs on future options for nuclear energy. For example, in January 2009, the U.S. Department of Energy (DOE) National Nuclear Security Administration (NNSA) released a draft Non-Proliferation Impact Assessment (NPIA) of the Global Nuclear Energy Partnership (GNEP) for public comment. [7] The draft NPIA analyzes the U.S. domestic nuclear fuel alternatives identified in the draft GNEP Programmatic Environmental Impact Statement (PEIS) for their potential impacts on the risk of nuclear proliferation and on U.S. non-proliferation goals. For details on the PEIS, see http://nuclear.gov/peis.html.

In evaluating the proliferation risk associated with the GNEP fuel cycle alternatives, the NPIA considered both policy and technical factors. [8] The policy evaluation drew on the relevant objectives of U.S. policy, which include discouraging the spread of enrichment and reprocessing technology, minimizing stocks of separated plutonium, promoting proliferation resistant technology, and improving international safeguards. The technical evaluation drew on the PR&PP methodology. [2] The draft NPIA concluded that recycling of spent fuel may offer opportunities for the United States to discourage the spread of enrichment and reprocessing technologies by participating in comprehensive nuclear fuel services. However, the NPIA also noted that, by separating relatively attractive materials from spent fuel, such recycling also involves new risks compared to the current once-through fuel cycle.

An Element of the Next Generation Safeguards Initiative (NGSI)

International safeguards are a central pillar of the nuclear non-proliferation regime. Administered by the International Atomic Energy Agency (IAEA), international safeguards serve to monitor nuclear activities under the Non-Proliferation Treaty (NPT) and are the primary vehicle for verifying compliance with peaceful use and nuclear non-proliferation undertakings.

The Department of Energy's NNSA undertook a broad review of international safeguards, which concluded that a comprehensive initiative to revitalize the international safeguards technology and human resource base by leveraging U.S. technical assets and partnerships was urgently needed to keep pace with demands and emerging safeguards challenges.

To address these challenges, NNSA launched the NGSI [9] to develop the policies, concepts, technologies, expertise, and infra–structure necessary to sustain the international safeguards system as its mission evolves over the next 25 years.

The deployment of new types of reactors and fuel cycle facilities, combined with the need to make the most effective and efficient use of limited safeguards resources, requires new concepts and approaches. The program plan for the NGSI calls for using the PR&PP methodology to evaluate new nuclear system designs for proliferation risk reduction. This will be helpful in establishing a global norm for designers to systematically identify tradeoffs and evaluate and compare different options. At the same time the methodology applications would have to be of sufficient quality to avoid unwarranted reductions in safeguards and physical protection efforts.

Safeguards by Design

There are ongoing and planned efforts both nationally [9] and internationally [10] to promote and implement the concept of safeguards by design (SBD) in the nuclear facility design process. These are very promising initiatives which can lead to effective and efficient introduction of safeguards early in the design process. Assessments of the benefits of SBD can be performed in the broader proliferation resistance framework. This is because, a gauge for how much proliferation risk reduction is being achieved in a SBD activity is needed to be able to understand its relative value with regard to economic, operational, safety, and security factors. An overarching PR&PP framework would help to guide effective and efficient safeguards in the design process.

Towards Harmonization with INPRO

In parallel with the multilateral effort by GIF PR&PP WG, and over the same time period, the International Atomic Energy Agency (IAEA) has been sponsoring development of an International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) to help to ensure that nuclear energy is available in the 21st century in a sustainable manner. See Pomeroy *et al.* [4] for additional information. In particular, INPRO has put forth basic principles, user requirements, and criteria for future nuclear energy systems, with similar broad goal areas to those that are being considered by GIF, including proliferation resistance and physical protection.

The INPRO approach [11] is primarily designed for nuclear energy system *users* (and thus guides the INPRO assessor in confirming that adequate proliferation resistance has been achieved in the nuclear energy system under consideration), but it can also give guidance to the *developer* of nuclear technology on how to improve proliferation resistance. The INPRO proliferation resistance approach identifies a *Basic Principle of Proliferation Resistance* and five *User Requirements* for meeting this Principle, along with seventeen indicators with specific criteria and acceptance limits.

The approaches share certain similarities, beginning with a common definition of proliferation resistance. Both approaches have a hierarchal analytical structure involving proliferation resistance principles, high-level evaluation factors and multiple measures or criteria related to each high-level factor. Both approaches treat proliferation resistance as a function of multiple *extrinsic measures* (*e.g.* safeguards, etc.) and *intrinsic features* (*e.g.* material attractiveness, etc.), and characterize proliferation resistance in terms of each. Both approaches recognize the concept of *barriers* to proliferation, but implement the concept differently. Neither approach aggregates its results into a single numerical value or grade, so that strengths and weaknesses under each of the main evaluation criteria are explicitly considered. Both approaches are primarily technical evaluations that incorporate institutional and policy contexts for the systems under consideration.

There are several notable differences between the two approaches. The INPRO approach focuses on the proliferation resistance of a declared, safeguarded nuclear energy system in a specific State, and implicitly excludes from the analysis clandestine facilities (including those that might be needed to complete a proliferation pathway) or a breakout scenario (in which a facility is *overtly* misused for proliferation purposes). In comparison, the GIF approach considers both declared and undeclared facilities and activities, to complete the proliferation pathway from acquisition and processing of material to fabrication of a nuclear explosive device as well as overt misuse following breakout.

IV. FUTURE DIRECTIONS

As the world increases its use and reliance on nuclear technologies for energy and other peaceful applications, there will be a need for a corresponding effort to assure that non–proliferation goals, as enunciated by the IAEA, are realized. There are many national and international programs that are aimed at providing this assurance. The PR&PP methodology is an analysis tool that can help to assess and manage the risks posed by threats to the peaceful use of nuclear technologies. Some area in which PR&PP studies could prove effective in reducing proliferation risk are indicated below.

As new and innovative design are developed for nuclear energy systems through GIF and INPRO, the PR&PP methodology approach will be essential to incorporating good design principles for proliferation resistance and physical protection into new emerging and viable concepts. The work that is just beginning between the PR&PP WG and the GIF SSCs will serve as a key model for how to implement this process. The PR&PP WG is in the early stages of planning a follow-on case study to the one recently completed on the example sodium fast reactor. Consideration is being given to a case study on a very high temperature gas-cooled reactor.

The PR&PP methodology approach can be a useful tool in developing safeguards by design as outlined in the Next Generation Safeguards Initiative and in recent parallel activities by the IAEA. Results of PR&PP evaluations can serve as clear discriminators among design alternatives and could thus help to make choices that reduce proliferation risk.

The PR&PP methodology can be used to evaluate the proliferation impacts associated with particular cases of export of nuclear fuel cycle technologies, materials, and information or to address the broader issue of evaluating the effectiveness of current practices.

V. CONCLUSION

The GIF PR&PP evaluation methodology was initially motivated by the need to have an approach to the assessment of new nuclear energy design concepts that were envisioned within the GIF program. The methodology that has been developed now enjoys wide international consensus and has been used in applications beyond the initial purpose. It is expected that subsequent applications of the methodology will 1) lead to refinement of the approach which will streamline and focus it to address issues of interest to end-users of the results and 2) have application to a more diverse set of applications that will enhance decision making in the PR&PP areas.

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References

1. DOE (U.S. Department of Energy), Nuclear Energy Research Advisory Committee and the Generation IV International Forum. December 2002. *A Technology Roadmap for Generation IV Nuclear Energy Systems.* GIF002-00, DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, Washington, D.C.

2. "*Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy System*" Revision 5, Generation IV International Forum, GIF/PRPPWG/2006/005, November 30, 2006; www.gen4.org/Technology/horizontal/PRPPEM.pdf. A summary of this methodology by P. Peterson appears in these Proceedings.

3. Panel Sessions Presentations on *Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems* at Global 2007 Conference, Sponsored by the American Nuclear Society, Boise, Idaho, September 9-13, 2007. Also, for the results of this Case Study, see these Proceedings of Global 2009, September 2009: M. Zentner *et al*. (Diversion Threat); G. Cojazzi *et al*. (Misuse Threat); J. Whitlock (Breakout Threat).

4. Pomeroy, G., E. Wonder, R. Bari, M. Zentner, "*Approaches to Evaluation of Proliferation Resistance of Nuclear Systems*" Proceedings of the Institute of Nuclear Materials Management, Annual Meeting, Nashville, TN, July 13-17, 2008.

5. Carre, F., S. Felix, "Proliferation Resistance and Physical Protection in the Generation IV International Forum System Design Concepts" Proceedings of Global 2009, September 2009.

6. Khalil, H., R. Bari, G-L. Fiorini, T. Leahy, P.F. Peterson, R. Versluis, "Integration of Safety and Reliability with Proliferation Resistance and Physical Protection for Generation IV Nuclear Energy Systems" Proceedings of Global 2009, September 2009.

7. Draft Non-Proliferation Impact Assessment (NPIA) of the Global Nuclear Energy Partnership (GNEP), December2008, *nnsa.**energy**.gov/**nuclear_nonproliferation**/documents/**GNEP_NPIA**.pdf*

8. Goodman, M.W., J.K. Sprinkle, A.M. Scheinman, "A Non-Proliferation Impact Assessment of the GNEP Alternatives," Proceedings of Global 2009, September 2009.

9. A description of this program can be found at http://nnsa.energy.gov/nuclear_nonproliferation/nuclear_safeguards.htm

10. "Facility Design and Plant Operation Features that Facilitate the Implementation of IAEA Safeguards" Workshop conducted from October 28-31, 2008 at IAEA Headquarters in Vienna, Austria. IAEA Report STR-360, February 2009.

11. International Atomic Energy Agency. Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems, INPRO Manual – Proliferation Resistance Volume 5 of the Final Report of Phase 1 of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1575, October 2007

RISK AND SAFETY WORKING GROUP: PERSPECTIVES, ACCOMPLISHMENTS AND ACTIVITIES

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I. INTRODUCTION

The Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) was created to promote a homogeneous and effective approach to assuring the safety of Generation IV nuclear energy systems. The six Generation IV reactor concepts that have been selected by the GIF members potentially present a diverse set of design and safety issues. A number of these issues differ significantly from those presented by the earlier generations of light water reactors. The overall success of the Generation IV program depends on developing, demonstrating, and deploying advanced system designs that exhibit excellent safety characteristics. While the RSWG recognizes the excellent safety record of nuclear power plants currently operating in GIF member countries, it believes that advanced technologies and a coherent safety approach in which safety is "built in, not added on" to the basic designs of nuclear systems hold the promise of making Generation IV energy systems even safer than the current generation of nuclear plants.

The Generation IV Technology Roadmap identifies three specific safety goals for Generation IV systems guides the Generation IV research and development program. The intent of the safety goals is to stimulate ideas for innovative energy systems that would achieve enhanced safety compared to that of the current plants, and to motivate and guide the research and development necessary to achieve that enhanced level of safety. These safety goals are:

1. *Generation IV nuclear energy systems will excel in safety and reliability.*
2. *Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*
3. *Generation IV nuclear energy systems will eliminate the need for offsite emergency response.*

The early work of the RSWG focused on defining a safety philosophy for Generation IV systems that is founded on lessons learned from current and prior generations of nuclear technologies, and on identifying the characteristics that may help achieve Generation IV safety goals. The RSWG is presently in the early stages of developing and demonstrating a methodology that will be used to assess and document the safety of Generation IV systems. This paper describes an integrated safety philosophy for Generation IV nuclear systems, desirable attributes to ensure safety, and the RSWG's early thinking about the integrated safety assessment methodology.

II. AN INTEGRATED PHILOSOPHY OF SAFETY

An effective and homogeneous approach to the safety of Generation IV systems must be based on a coherent and well-founded safety philosophy. In its work to date, the RSWG has recommended that the following postulates should underlie such a safety philosophy:

- **Opportunities exist to further improve on nuclear power's already excellent safety record in most countries.** As a starting point, the RSWG recognizes that the level of safety that has been attained by the vast majority of operating nuclear power plants (Generation II) in most countries of the world is already very good. Relative to Generation II systems, applicable quantitative safety objectives for third generation (*e.g.* AP1000 and EPR) nuclear power plants are very ambitious and provide a further improved level of safety. The RSWG believes that, although not formally required, further enhancement in the level of safety associated with Generation IV technologies is possible. Such improvements can be realized through advanced technologies and early application of an integrated safety approach driven by an assessment methodology that helps identify improvements to the developing design. Such improvements will focus on safety provisions that will be ***"built-in"*** to the fundamental design rather than ***"added on"*** to the system architecture.

- **Safety improvements should simulta–neously be based on several elements which will require specific R&D efforts**. These include the notion of "optimal risk reduction"; the adoption of ambitious safety objectives that will drive the research required to attain those objectives; the application of innovative technologies; an emphasis on accident prevention backed up by mitigation; the development of robust safety architecture; and improved means of demonstrating the system's safety robustness.

- **The diversity of the Generation IV systems and the need for a homogeneous strategy applicable for the design and the assessment of these systems justify an updated safety approach.** The traditional approach to safety is one that has consisted largely of prescriptive requirements based largely on "engineering judgment". The notion of the "design basis accident" as a bounding case underlies much of the historical safety basis for nuclear plants that began operation in the sixties and seventies. Advancements and analytical methods developed since then support an updated safety approach. Such an approach must include formal consideration of risk and safety issues throughout the design process, and must provide for prevention and mitigation relative to a broad spectrum of potential accident initiators and conditions.

- **The principle of "defense in depth" has served the nuclear power industry well, and must be preserved in the design of Generation IV systems.** Defense in depth is the key to achieve safety robustness, thereby helping to ensure that Generation IV systems do not exhibit any particularly dominant risk vulnerability. Embodied within the principle of defense in depth is the notion that safety margins must exist as an effective response to uncertainty.

- **The Generation IV design process should be driven by a "risk-informed" approach.** The RSWG believes that safety and economics of Generation IV designs can be positively impacted by formally adopting, as a complement of the deterministic approach, the use of PSA techniques and complementary tools as design drivers throughout the design process.

III. DESIGN AND ASSESSMENT OF INNOVATIVE SYSTEMS

Specific details of Generation IV systems designs must, of course, be left to their respective design teams. The RSWG, therefore, does not offer prescriptive guidance with respect to design issues. Rather, the RSWG has worked to define certain general design attributes or criteria that are believed to offer benefit in terms of helping to achieve the safety goals for Generation IV systems. Some of these attributes include:

- **The Design Basis for Generation IV energy systems should cover the full range of safety significant conditions.** The historical notion of a single bounding design basis accident must be replaced by a "spectrum" of possible accidents that, while of low probability, represents with high confidence the range of physical events and phenomenology that could conceivably challenge the plant. Specific efforts, both analytical and empirical, should be made for demonstrating the "practical elimination" of initiators, sequences or situations associated with the extremely low residual risk.

- **Objectives and practices for design improvement must be explicit and complementary**. To efficiently establish these practices, **four complementary ways** should be followed by the designer: 1) critical and systematic examination and consideration of feedback from experience; 2) full implementation of the concept of defense in depth in an effective and measurable manner; 3) rationalization of the design approach by the deliberate adoption of the ALARP principle on a cost benefit basis; 4) special attention should be devoted to the reinforced treatment of the severe plant conditions through provisions of measures that provide defense (*i.e.*, prevention and mitigation) against such conditions.

- **The demonstration of a concept's safety robustness rests on the capacity of the designer and the developer to demon–strate and to guarantee exhaustiveness in the recognition of risks stemming from phenomena considered for the design.** Whenever possible, plant design features based on natural phenomena and physical properties of materials should be used to demonstrate in an "intuitive" way the ability of the plant to arrest the accident progression. This must be done with an adequate degree of confidence, based on an understanding of the associated uncertainties and the provision of sufficient safety margins in response to those uncertainties.

IV. A METHODOLOGY FOR ASSESSING AND DOCUMENTING THE SAFETY OF GENERATION IV SYSTEMS

One principal focus of the RSWG's charter is the development and demonstration of an integrated methodology that can be used to assess and document the safety of Generation IV nuclear systems. Although the RSWG is still in the very early stages of developing and presenting such a methodology, this activity is the current focus of the Group, and the elements of the methodology have been largely defined. The methodology is tentatively called the Integrated Safety Assessment Methodology (ISAM).

It is envisioned that the ISAM will be used in three principal ways:

- The ISAM is intended for use throughout the concept development and design phases with insights derived from the ISAM serving to actively drive the course of the design evolution. In this application of the methodology, the ISAM is used to develop a more detailed understanding of design vulnerabilities, and resulting contributions to risk. Based on this detailed understanding of vulnerabilities, new safety provisions or other design improvements can be identified, developed, and implemented relatively early.

- Selected elements of the methodology will be applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other issues that are important to decision makers.

- The ISAM can be applied in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria. In this way, the ISAM will allow evaluation of a particular Generation IV concept or design relative to various potentially applicable safety metrics or "figures of merit". This *post facto* application of the ISAM will be especially useful for regulators and other decision makers who require objective measures of safety for licensing purposes, or to support certain late-stage design selection decisions.

It is specifically NOT intended that the methodology be used to dictate design requirements, that it dictate compliance with quantitative safety goals, or that it in any other way constrains designers. The sole intent is to provide a useful methodology that contributes to the attainment of Generation IV safety objectives, that yields useful insights into the nature of safety and risk of Generation IV systems, and that permits meaningful evaluations of Generation IV concepts with respect to safety.

Attributes of an Effective Safety Assessment Methodology

In formulating a Generation IV safety assessment methodology, the RSWG has sought to incorporate the following attributes:

- The methodology should consist of, or be largely based on existing tools that are widely accepted for their validity. Thus, the methodology should minimize the need for developing new tools and the potentially lengthy period of validation that may be necessary. When necessary, however, the methodology must support incorporation of new analysis techniques to address issues or phenomena specific to advanced energy systems or demonstration of the robustness of those systems.

- The methodology must be comprehensive, understandable, user-friendly, and efficient.

- The methodology must allow for the integration of a diverse range of multi-disciplinary inputs including those that are primarily probabilistic and those that are primarily deterministic in nature, as well as those that are principally qualitative and those that are principally quantitative.

- Based on the desirability of offering a graded approach to technical issues of varying complexity and importance, practicality and flexibility must be reflected in the methodology.

- Throughout the development process, the safety assessment methodology must help designers understand design vulnerabilities, and how alternative design solutions can reduce or eliminate those vulnerabilities. In order to successfully fulfil this role, the methodology must yield information about which aspects of design contribute most to the level of risk associated with that concept or design. Thus, the methodology must serve to do more than just measure safety after the design is complete. *The methodology must actively contribute to the development of designs that fulfil the safety objectives of Generation IV systems.*

- Importantly, the methodology must provide information that permits an understanding of the level of uncertainty associated with the measured level of safety, as well as an understanding of the sources of that uncertainty.

- Based largely, but not exclusively, on a systematic understanding of sources and magnitudes of uncertainties, the metho–dology must help identify areas for additional research, data collection, and improved analytical models.

- Within a given concept, the methodology must support comparisons of potential alternative design options.

- The methodology must yield information that allows comparison of a concept or design relative to established safety metrics or "figures of merit."

- The methodology must yield a mix of both qualitative and quantitative information that will support eventual licensing and regulatory processes.

- To the extent that is appropriate, the methodology should be consistent with other relevant guidance and documentation including the RSWG Safety Philosophy document (Ref. 1), the PRPP methodology (Ref. 7), and other work including the US NRC NUREG-1860 (Ref. 2), the IAEA TECDOC-1570 (Ref. 3), and others.

ISAM Overview

The ISAM provides an integrated set of tools that reasonably fulfils the list of desired methodological attributes outlined above. Although the ISAM is fundamentally based on PSA, the integrated methodology consists of five distinct analytical tools. It is intended that each tool be used to answer specific kinds of safety-related questions in differing degrees of detail, and at different stages of design maturity. By providing specific tools to examine relevant safety issues at different points in the design evolution, the ISAM as a whole offers the flexibility to allow a graded approach to the analysis of technical issues of varying complexity and importance. The methodology is well integrated, as evidenced by the fact that the results of each analysis tool support or relate to inputs or outputs of other tools. Although individual analytical tools can be selected for individual and exclusive use, the full value of the integrated methodology is derived from using each tool, in an iterative fashion and in combination with the others, throughout the development cycle.

Because the development of the methodology is still in its very early phases, all information concerning the methodology should be regarded as tentative, preliminary, and pre-decisional. At the current time, the RSWG believes that the ISAM will consist of the following major elements:

- Qualitative Safety Features Review (QSR)

The Qualitative Safety Features Review is a new tool that provides a systematic means of ensuring and documenting that the evolving Generation IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed in the RSWG's first report entitled, "Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems." Although this element of the ISAM is offered as an optional step, it is believed that the QSR provides a useful means of shaping designers' approaches to their work to help ensure that safety truly is "built-in, not added-onto" since the early phases of the design of Generation IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of "defense in depth" high safety reliability, minimization of sensitivity to human error, and other important safety characteristics might best be incorporated. The QSR is not regarded as a tool that allows an analyst to determine whether or not a developing concept is "good enough", but rather, provides a measure of discipline to help ensure that certain desirable characteristics are incorporated into the design in its earliest phases. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues that will be analyzed in more depth in those other analytical steps.

- Phenomena Identification and Ranking Table (PIRT)

The Phenomena Identification and Ranking Table is a technique that has been widely applied in both nuclear and non-nuclear applications. The PIRT provides a structured means of identifying and analyzing a wide variety of off-normal scenarios that potentially challenge the viability of complex technological systems. As applied to Generation IV nuclear systems, the PIRT is used to identify a spectrum of safety-related scenarios or phenomena that could affect those systems, and to rank order those scenarios on the basis of their frequencies, their potential consequences, and state of knowledge related to associate phenomena (*i.e.*, sources and magnitudes of phenomenological uncertainties).

The PIRT is used initially in the pre-conceptual design phase of a system's development, and is applied iteratively throughout the development process. It is to be used as an early screening tool to identify, categorize, and characterize phenomena and issues that are potentially important to risk and safety of a Generation IV system. The PIRT can be focused on very general issues, or on highly specific design issues, depending on the need. The method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools that comprise the ISAM. As such, the PIRT forms an input to both the Objective Provision Tree (OPT) analyses, and the Probabilistic Safety Analysis (PSA) in identifying mechanisms and initiating events which will challenge the safety functions. In the case of the PSA, the PIRT is particularly helpful in defining the course of accident sequences, and defining safety system success criteria. The PIRT is essential in helping to identify areas in which additional research may be helpful to reduce uncertainties.

- Objective Provision Tree (OPT)

The Objective Provision Tree is a relatively new analytical tool that is enjoying increasing use. The International Atomic Energy Agency (IAEA) has been a particularly influential developer and proponent of this analysis tool. The purpose of the OPT is to ensure and document the provision of essential "lines of protection" to ensure successful prevention or mitigation of phenomena that could potentially damage the nuclear system. There is a natural interface between the OPT and the PIRT in that the PIRT identifies phenomena and issues that could potentially be important to safety, and the OPT focuses on identifying design provisions intended to prevent, control, or mitigate the consequences of those phenomena.

The OPT can be applied early in the pre-conceptual design phase, and iteratively through conceptual design. Note that the OPT is an entirely qualitative analysis method. As such, its purpose is to inform the design process and to help structure inputs that will eventually make their way into the PSA. The OPT can be extremely useful in helping to focus and structure the analyst's understanding of accident sequence phenomenology, sequence success criteria, and related issues. It will help providing the right requirements (*e.g.* requested performances and reliability) for the design of the implemented provisions.

- Deterministic and Phenomenological Analyses (DPA)

Classical Deterministic and Phenomenological Analyses, including thermal-hydraulic analyses, computational fluid dynamics (CFD) analyses, reactor physics analyses, accident simulation, materials behavior models, structural analysis models, and the like collectively constitute a vital part of the overall Generation IV ISAM. These traditional deterministic analyses will be used as needed to understand a wide range of safety issues that must guide concept and design development, and will form inputs into the PSA. These analyses typically involve the use of familiar deterministic safety analysis codes. It is anticipated that DPA will be used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

- Probabilistic Safety Analysis (PSA)

PSA has been widely used in a variety of nuclear and non-nuclear applications since the early 1970s. As a widely accepted, integrative method that is rigorous, disciplined, and systematic, PSA forms the principal basis of the ISAM. PSA can only be meaningfully applied to a design that has reached a sufficient level of maturity and detail. Thus, PSA is to be performed, and iterated, beginning in the late pre-conceptual design phase, and continuing through the final design stages addressing licensing and regulation concerns. In fact, as the concept of the "living PSA" (one that is frequently updated to reflect changes in design, system configuration, and operating procedures) is becoming increasingly accepted, the RSWG is advocating the idea of applying PSA as the earliest practical point in the design process, and continuing to use it as a key decision tool throughout the life of the plant or system. Although the other elements of the

ISAM have significant value as stand-alone analysis methods, to a significant degree, their value is enhanced by the fact that they serve as useful tools in helping to prepare for, and to shape, the PSA once the design has matured to a point where the PSA can be successfully applied.

Fundamentally, the PSA provides a structured means of identifying the answers to three basic questions related to the safety of Generation IV systems. These are:

- What can go wrong?
- How likely is it?
- What are the consequences?

The centerpiece of the ISAM is a "full scope" PSA that considers both internal and external events and models potential accident phenomena from the hypothetical occurrence of an initiating event through the point at which accident progression is either arrested, or offsite consequences are realized.

One of the key strengths of the PSA is that it facilitates a systematic understanding of the uncertainties relating to the safety (or risk) of a Generation IV system. Uncertainties arise from a number of sources. The traditional response to these safety-related uncertainties has been the provision of additional "safety margin" in the design, often based largely on "engineering judgment", to provide assurance that in the event of any accident, severe loss or damage will not occur. Adding such safety margins is, of course, expensive, and may also lead to an inappropriate focus on some aspects of design and operation to the detriment of other issues that may, in fact, be more important to safety. By facilitating a disciplined, systematic understanding of the sources and magnitudes of safety-related uncertainties, the PSA will play a key role in helping to ensure that cost and safety issues are more optimally balanced.

V. CONCLUSION

Advanced technologies and a safety approach driven by insights derived from an integrated safety assessment methodology hold the promise of making Generation IV energy systems even safer than the current generation of nuclear plants.

The ISAM is best thought of as a toolkit of useful analysis tools. Although the ISAM is essentially a PSA-based safety assessment methodology for Generation IV systems, the strength of the ISAM is that it offers tools that are tailored to answering specific types of questions at various stages of design development, and that the elements of the methodology complement and support one another in a way that contributes to a much more complete understanding of the range of safety issues. It is anticipated that using the elements of the ISAM in an integrated way will result in optimizing safety, reducing technology development cycle time, reducing development costs, and facilitating licensing of Generation IV systems.

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References

1. "*Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems,*" Rev. 1, Generation IV International Forum, GIF/RSWG/2007/002, November 24, 2008.
2. "*Feasibility Study for a Risk-informed and Performance-Based Regulatory Structure for Future Plant Licensing* (NUREG 1860)." US Nuclear Regulatory Commission, December 2007.
3. "*Proposal for a Technology-Neutral Safety Approach for New Reactor Designs* (TECDOC-1570)." International Atomic Energy Agency, September 2007.
4. "*Determining the Quality of Probabilistic Safety Assessment (PSA) for Applications in Nuclear Power Plants* (TECDOC-1511)." International Atomic Energy Agency, July 2006.
5. "*A Risk-Informed, Performance-Based Regulatory Framework for Power Reactors* (NEI-02-02)." Nuclear Energy Institute, May 2002.
6. "*Policy Statement on the Regulation of Advanced Reactor.*" NRC-2008-0237, US Nuclear Regulatory Commission, Federal Register. October 14, 2008.
7. "*Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems,*" Rev. 5, Generation IV International Forum, GIF/PRPPWG/2006/005, November 30, 2006.

GAS REACTORS – A REVIEW OF THE PAST, AN OVERVIEW OF THE PRESENT AND A VIEW OF THE FUTURE

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Abstract

Gas-cooled reactors have a rich history and a promising future. They were among the first nuclear plants to be commercially deployed and are now the subject of revitalized interest for future deployment. Gas-cooled reactors may be critical to the management of climate change and energy security in the coming decades. This paper reviews the history of gas-cooled reactors from the carbon dioxide (CO_2) cooled reactors in the United Kingdom and France to the prototype helium cooled reactors in the United States and Germany. The paper summarizes the current research and development (R&D) work supported by the Generation IV International Forum (GIF) Gas Cooled Reactor Program in general and discusses the on-going gas reactor R&D and demonstration work in the United States, People's Republic of China, and Republic of South Africa. Finally, the paper summarizes the broad range of potential applications of high temperature gas-cooled reactors including electricity generation, process heat production, unconventional hydrocarbon development, and hydrogen generation.

I. INTRODUCTION

This paper documents the evolving applications of gas-cooled reactors; past, present and future. An overview of the past covers the experience with commercial gas-cooled reactors to date. An assessment of the present focuses on the technical work coordinated by GIF to support the future deployment of the Very-High Temperature Reactor (VHTR) and associated national programs to support nearer term deployment of High-Temperature Reactors (HTR). A forecast of the future describes the potential applications of high-temperature gas cooled reactors in a world that is concerned with global climate change and the utilization of scarce resources. The gas reactor has a rich history and a promising future.

Gas-cooled reactors were among the first nuclear plants to be commercialized from military applications, which used natural uranium to produce plutonium for nuclear devices. The temperature of the core outlet has risen steadily from the earliest CO_2 reactor (340°C) to nearly 1 000°C. This increase in temperature will enable the application of gas-cooled reactors to expand from the generation of electricity to the production of process heat, and the production of nonconventional hydrocarbons, and ultimately, to the production of hydrogen.

II. EXPERIENCE WITH GAS-COOLED REACTORS

II.A. CO_2 Cooled Reactors

The earliest commercial gas-cooled reactors were derived from plutonium production reactors. They used natural uranium, thus not requiring expensive enrichment facilities. They also used CO_2 as a coolant and graphite as a moderator. The United Kingdom and France were the primary developers and users of the technology. Gas reactors still provide most of the UK electricity generated with nuclear energy, which is almost 20% of the UK's total electricity. France has decommissioned all of its gas-cooled reactors and replaced them with light water reactors. The "lessons learned" from the deployment of the CO_2 reactors are discussed in the following sections.

II.A.1 United Kingdom

II.A.1.a. Magnox Reactors

The earliest gas-cooled reactors in the UK were called Magnox reactors, because they utilized magnesium (Mg) as the cladding for the fuel. The first Magnox plant was the Calder Hall station with four 220 MWt (51 MWe) reactors. Calder Hall operated from 1956 to 2003 with very good performance. The CO_2 coolant left the core at 340°C and 0.66 MPa pressure. There were ten additional Magnox stations. The Magnox reactors represented almost 4.3 GWe of generation. The average station life was almost 40 years. Two Magnox stations remain in operation. In general, the Magnox plants performed very well for the UK. The life-time fleet average capacity factor was 70.3%. Similar reactors were deployed in Italy and Japan.

II.A.1.b. Advanced Gas Reactors (AGR)

In order to improve the performance of the Magnox plants, the UK developed a second generation CO_2 cooled design, the advanced gas reactor. The increase in thermal power level resulted in the core outlet temperature increasing to 640°C and pressure to 4.0 MPa, the temperature being limited by the chemical activity of the CO_2. The resulting fuel temperature required the use of stainless steel cladding, and the uranium in the fuel had to be enriched to between 2.5 and 3.5% uranium 235 (U^{235}). AGRs with almost 8.4 GWe of capacity were built at six sites in the UK.

The overall performance of the AGRs has been adequate. The on-line refueling feature had to be abandoned due to unacceptable vibrations. Graphite moderator blocks experienced cracking. Boiler issues at Hinkley Point and Hunterston have limited the output of the plants to 70% normal capacity. Following the AGR construction, the CEGB elected to pursue light water technology for Sizewell B and subsequent nuclear stations.

II.A.2. France

France initially followed a similar path to the UK in their development of CO_2 cooled, natural uranium reactors. The initial designs were used by the Commissariat à l'Energie Atomique of France (CEA) for dual purposes to produce plutonium for the French weapons program and power for the grid. Subsequently, Electricité de France (EdF) owned and operated six gas-cooled power reactors with a generating capacity in excess of 2.2 GWe. The natural uranium fuel was clad in Mg/zirconium alloy. The core outlet temperatures reached 385°C with a 2.45 MPa system pressure in the later reactors.

All of the French reactors were shut down prior to end-of-life for economic reasons coupled with the long-term transition to light water technology that began in 1972. The St. Laurent plants had numerous problems with steam generator tube leaks. The condition was finally mitigated by changes made in the Bugey 1 plant.

The Spanish built a gas-cooled reactor at Vandellos based upon the St. Laurent A design. This reactor was shut down prematurely due to a major turbine-generator fire.

II.B. Helium Cooled Reactors

The core outlet temperature in gas-cooled reactors can be increased above the limit imposed by the CO_2 chemical attack of the graphite if helium is used as the coolant. Helium is much more expensive and has better heat transfer properties than CO_2. Helium is much more demanding from a system leak tightness perspective due to its small molecular size. Helium is the coolant of choice for future gas cooled reactors.

II.B.1.United Kingdom

One of the earliest developers of the high temperature gas-cooled reactors using helium as a coolant was the UK's Atomic Energy Authority (UKAEA). As part of an Organization for Economic Co-operation and Development (OECD) project, the UKAEA developed and built the Dragon demonstration reactor. The Dragon reactor was a 20 MWt helium cooled, graphite-moderated reactor with a core outlet temperature of 750°C and pressure of 2.06 MPa. The reactor operated from 1959 to 1976. While the reactor had no means for producing electricity, it was a valuable demonstration for the use of helium as a coolant. The UK did not pursue the use of helium beyond the Dragon reactor.

II.B.2. United States

There were two major demonstration plants built and operated in the United States. The first was Peach Bottom Unit 1, followed by the larger Ft. Saint Vrain (FSV) Plant. A brief discussion of each follows.

II.B.2.a. Peach Bottom Unit 1

The Peach Bottom Unit 1 reactor was a high-temperature gas-cooled reactor designed and built by the General Atomics Company (GA) for the Philadelphia Electric Company. The reactor used helium as a coolant and graphite as a moderator. The thermal power level was 110 MWt and 48 MWe. The core outlet temperature was 794°C and pressure was 2.4 MPa. The reactor vessel was made of steel, and there were two core designs. The first core used coated fuel particles of U-235 and Thorium-232 carbide with a single layer of anisotropic carbon. Fast neutron-induced dimensional changes cracked 90 out of the 804 sleeves containing the fuel particles. The sleeve cracking did not impair reactor operation, and the coolant activity was less than 7% of the design level. The second core incorporated buffered isotropic pyrolytic carbon (BISO) fuel particles. The second core operated its full design life with no fuel sleeve failures and one-millionth the design coolant activity. The plant went into commercial operation in mid-1967, and retired for decommissioning in October 1974. The shutdown was a planned economic decision. The overall capacity factor over its life was an impressive 74%, and the plant was available 88% of the time.

Peach Bottom 1 provided several important outcomes including excellent agree–ment between the design calculations and the actual performance, excellent fuel performance of the BISO fuel, and excellent steam generator performance and load following capability.

II.B.2.b. Fort Saint Vrain

The FSV reactor was part of the U.S. Atomic Energy Commission (USAEC) Power Reactor Demonstration Program with most of the funding coming from the owner/operator, Public Service of Colorado. While the FSV reactor was designed by GA, it was very different from the Peach Bottom Unit 1 design. FSV used helium as the coolant and graphite as the moderator. The power output was much greater, 842 MWt and 330 MWe, the core outlet temperature was 778°C, and the system pressure was increased to 4.83 MPa. The reactor vessel was reinforced concrete, and the core was comprised of fuel compacts containing three layer (TRISO) fuel particles in graphite blocks. The operating license was issued by the USAEC in 1973; and full power was reached in November 1981. FSV was shut down in 1989 due to financial reasons, as the operating and maintenance costs exceeded the plant revenues. The overall capacity factor for FSV was less than 30%.

There were numerous problems with the FSV. These problems included core outlet temperature fluctuations (fixed by adding core restraints); leakage of water into the core from the helium circulator water bearings; steam generator leaks and header cracks discovered at end-of-life; reserve shutdown system mal-function, emergency pump cavitation (one-year delay); hot helium bypass on control rod drives, and a hot spot on the core support floor. In spite of the problems, there were a number of valuable lessons learned from the design, construction and operation of FSV. These positive lessons included a much lower fission product release (than expected); an excellent agreement between calculations and actual performance; the computer control of fuel handling worked well; the helium purification system worked well; and the reinforced concrete reactor vessel performed well and systems, in general, performed as designed.

GA received orders for ten, large commercial plants, which were cancelled in the early 1970s due to a combination of the oil embargo and reduced electricity demand. The commercial designs were larger, improved versions of the FSV reactor. GA withdrew from the commercial nuclear business in 1975.

II.B.3. Germany

The high-temperature gas reactor program in Germany was similar to the U.S. program in that it pursued high temperatures using helium as a coolant and graphite as a moderator. The primary difference was in the fuel configurations. The United States used fixed graphite blocks with fuel compacts containing coated fuel particles. The German program used a mobile pebble fuel configuration that permitted on-line refueling. The pebbles contained similar coated particle. There were two major demonstrations of the technology, the Arbeits-gemeinschaft Versuch Reaktor (AVR) and the Thorium High Temperature Reactor (THTR).

II.B.3.a. The Arbeitsgemeinschaft Versuch Reaktor (AVR)

The AVR was a Federal Republic of Germany project at the Julich Research Center designed to demonstrate the feasibility of using spherical fuel elements (pebbles) and high temperatures. The AVR operated from 1967 to the end of 1988, when it was closed in response to the political pressures raised by the Chernobyl nuclear accident. The AVR had a thermal output of 49 MWt and an electrical output of 15 MWe. The AVR used a steel reactor vessel. The initial core outlet temperature was 850°C, which was subsequently raised to 950°C in 1974. The system pressure was 1.1 MPa. Several fuel designs were used at the AVR. The initial fuel did not perform as well as expected. The three-layer TRISO fuel was ultimately used with very good success. In spite of a major repair outage to repair damage from a steam generator leak in 1978, the AVR returned to service in 1980 and achieved a respectful 66.4% overall availability.

There were a number of lesson learned from the AVR experience. These include pebble bed reactors work; Light Water Reactor (LWR)-type containments are not required for future high-temperature gas-cooled reactors; modular reactors are feasible. In addition, success was demonstrated in reactor operations (normal, transient and accident conditions), materials (fuel, graphite, ceramics, metallics), design (control, vessel and auxiliary systems), and pebble bed fuel manufacture and handling.

The AVR technology was transferred to South Africa in the late 1990s and became the basis for the Pebble Bed Modular Reactor. The technology was also transferred to China, where it became the basis for the High-Temperature Reactor (HTR) Program.

II.B.3.b. Thorium High Temperature Reactor (THTR)

The second German pebble bed reactor was similar in power level to the FSV and was intended to be a building block for the German high-temperature gas-cooled reactor program to achieve commercial scale power plants. The plant thermal output was 750 MWt and 300 MWe. The core outlet temperature was 750°C, and the system pressure was 3.9 MPa. Like the FSV, the THTR used a reinforced concrete reactor vessel with integral cooling circuit. The construction of THTR began in 1971 but was not completed and licensed until 1984. The THTR was shut down in 1989 in part due a shortfall in funding and also in response to the political pressures raised by the Chernobyl nuclear accident

The lessons learned from the operation of the THTR include plant maintenance workers encountered very low doses of radiation; control rods can be safely inserted into a bed of fuel pebbles, and reliable on-line refueling and pebble discharge systems can be designed and operated. Finally, as for all reactors, licensing delays can result in major redesign and costly delays.

II.B.4. Japan

The High Temperature Engineering Test Reactor (HTTR) is the center piece of the Japanese high-temperature gas-cooled reactor program. The HTTR is a 30 MWt prismatic block with outlet temperatures as high as 950°C. The HTTR construction began in 1990, and criticality was achieved in 1998 with very good operating experience. The HTTR was designed to establish gas reactor technology and nuclear heat utilization technology including the production of hydrogen using the sulfur iodine process.

II.B.5. People's Republic of China

The Chinese high-temperature gas-cooled reactor program is based on the pebble bed design imported from Germany. The center piece of the Chinese program is the 10 MWt test reactor called the HTR-10. The HTR-10 is 10 MWt pebble bed reactor with an outlet temperature of 700°C (up to 900°C) and a system pressure of 3 MPa. The reactor construction began in 2000, and full power operation was achieved in 2003. The test reactor has performed well and significant safety tests demonstrate the passive cooling capability of the HTR. Their program also includes the HTR-PM, a commercial prototype presently under design and construction.

III CURRENT DEVELOPMENT AREAS

III.A. The GIF Gas-Cooled Reactor Program

The GIF Program encompasses the development of advanced reactors and fuel cycles to support the broader deployment of nuclear energy to help reduce greenhouse gas emissions and increase global energy security. One of the advanced reactors selected by GIF is the very high-temperature reactor (VHTR). The VHTR is defined as a helium-cooled, graphite moderated reactor with a core outlet temperature in excess of 900°C and a long-term goal of achieving an outlet temperature of 1 000°C. The VHTR is suited for a broad range of applications, including the production of hydrogen from water. Members of the GIF VHTR System Arrangement include the United States, France, Japan, United Kingdom, People's Republic of China, the Republic of Korea, Canada, Switzerland, and EURATOM.

In order to achieve the ambitious goals of the VHTR, GIF has established a research plan. The main R&D areas of the VHTR System Research Plan are briefly summarized below.

III.A.1. Computational Methods Development and Validation

Computational methods development and validation are major activities for the assessment of the reactor performance, in normal, incidental and accidental conditions. Computational tools are needed in areas such as thermal-hydraulics, thermal mechanics, core physics, chemical transport, and the derivate couplings. Numerical models will be specifically developed and validated to meet the pebble bed and the prismatic type core reactors requirements. Extension and validation of existing engineering and safety analysis methods are especially required to yield new design and safety approaches, new materials, operating regimes, and component configuration in the models.

Code calculations will be validated through benchmark tests and code-to-code comparisons from basic phenomena to integrated experiments, supported by HTTR (30 MW) tests, or HTR-10 (10 MW) tests or by past technology high temperature reactor data (*e.g.* AVR, Fort St Vrain, etc.).

III.A.2. Fuel and Fuel cycle

TRISO coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions. Furthermore, the standard design using uranium dioxide can evolve along with the improvement of its performance through the use of a uranium oxycarbide fuel kernel or a zirconium-carbide coating for enhanced burn-up capability, reduced fission product permeation and increased resistance to core heat-up accidents (above

1 600°C). The research will include fuel characterization, post-irradiation examination, safety testing, and fission product release evaluation as well as chemical and thermo-mechanical materials properties in representative conditions.

Fuel cycle back-end R&D will encompass spent fuel treatment and disposal, as well as used graphite management. A “once through” cycle is initially envisioned. However, the potential for deep-burning of plutonium and minor actinides in a VHTR, and the use of thorium-based fuel will be included in future R&D as important steps toward a closed cycle.

III.A.3. Materials

Reliable materials performance is key to the viability of the VHTR. The projected R&D for improved materials includes materials development and qualification; development of design codes and standards; improved manufacturing, installation, and construction techniques for key components. The service temperatures range from a near-term core outlet temperature between 750 and 900°C, for which existing materials may be used. The longer-term goal of 1 000°C requires the development and qualification of new materials.

The materials of particular interest include:

- Graphite for the reactor core and internals.
- High-temperature metallic materials for internals, piping, circulators, valves, heat exchangers, steam generators, gas turbine sub-components.
- Ceramics and composites (C-C, SiC-SiC, etc.) for control rod cladding and other specific reactor internals, as well as for advanced intermediate heat exchangers and gas turbine components for very-high temperature conditions.

III.A.4. Components and High Performance Turbomachinery

Design and construction investigations will address key components of the VHTR system such as the reactor pressure vessel, core, internals, circulators, valves, hot duct and heat exchangers, reactor cavity cooling system (RCCS) and other subsystems. Highly efficient generation of electricity with a VHTR requires a closed Brayton power conversion system. Anticipated R&D tasks include gas turbine and compressor system design and manufacturing, rotor dynamics, magnetic bearing technology, system layout, maintainability, and control system. In conjunction with these above efforts, new welding techniques shall be developed, and dedicated test loops will be needed to support the component design work

III.A.5 Hydrogen Production and Other Process Heat Applications

The principal candidates for hydrogen production from water are (1) the sulphur/iodine (S/I) thermo-chemical cycle and (2) the high-temperature electrolysis (HTE) process. Integrated test loops will help assess the performance and optimize the processes prior to building a demonstration scale prototype. Such test loops will assist the development heat exchange and transport components.

Coupling the hydrogen process technology with the nuclear reactor is another key element in the VHTR development. Considerations include interfacing events between nuclear and nonnuclear plants, areas of particular interest include thermal load management, hydrogen fires and explosions, toxic and hazardous material releases, tritium permeation and thermal disturbances caused by the hydrogen production system transients.

Additional process heat applications for the VHTR are extremely important to both energy security and global climate change management. These applications are discussed in more detail in Section IV of this paper.

III.B The U.S. Gas Cooled Reactor Program

In the United States, high-temperature gas-cooled reactor (HTR) development work is funded by the U.S. Department of Energy’s (DOE) Office of Nuclear Energy (NE) Generation IV Program. Two tracks are being pursued. The first is to support the technologies required for near-term commercialization, the second is to extend the capabilities into even higher temperature regimes. The reference near-term concept is a helium-cooled, graphite-moderated, thermal neutron spectrum reactor with an outlet temperature of 750 to 850°C. The reactor core configuration may be either a prismatic graphite block or pebble bed. These near-term concepts have the potential to extend the benefits of nuclear energy beyond the electrical grid by providing industry with carbon-free, high temperature process heat for a variety of applications including petroleum refining, bio-fuels production, and production of feedstock for use in the fertilizer and chemical industries. The reactor thermal power and core configuration will be designed to assure passive decay heat removal without fuel damage during any potential accident. An integral part of the U.S. Generation IV VHTR Program is the development of a regulatory framework with the U.S. Nuclear Regulatory Commission

The U.S. Generation IV VHTR R&D activities are closely integrated with the GIF VHTR activities. Key aspects of the U.S. Generation IV VHTR R&D are discussed below.

III.B.1 Fuels

The U.S. Generation IV VHTR's AGR Fuel Development and Qualification Program are designed to provide a fuel qualification baseline to support regulatory acceptance. The AGR Fuel Development and Qualification Program supports the near-term deployment of gas reactor technology by reducing market entry risks posed by technical uncertainties associated with fuel production and performance.

The program is: (1) developing techno–logies for the manufacture of very high-quality fuel kernels, TRISO-coated particles, and compacts; (2) irradiating fuel to high burnup at prototypical powers; (3) testing the irradiated fuel during worst-case accident simulations, and (4) developing and validating physically based computer models of the fuel and fission product transport behaviour.

III.B.2 Materials

The VHTR Materials R&D Program is testing and qualifying the key materials commonly used in very high-temperature designs. The materials R&D Program encompasses the materials needed for the VHTR reactor system, power conversion unit, intermediate heat exchanger, and associated balance of plant. The order of priority for the VHTR materials R&D is as follows:

- Test and qualify core graphite materials.
- Develop an improved high-temperature design methodology for use of selected metals at very high temperatures.
- Develop American Society of Mechanical Engineers (ASME) and American Society for Testing and Materials (ASTM) codes and standards.
- Perform environmental testing and thermal aging of selected hightemperature metals.
- Irradiate, test, and qualify reactor pressure vessel (RPV) materials.
- Develop and qualify composites for use in control rod cladding and guide tubes.
- Resolve RPV fabrication and transporta–tion issues.

III.B.3.Computational Methods

Included in the U.S. Generation IV VHTR R&D effort is the advancement of analytical methods and modelling to support gas reactor design including establishing qualification and validation criteria and experiments. The methods efforts will develop improved analytical codes and validate applications of these codes using data from scaled experiments and prior experience. A major focus will be on the development of tools to assess the reactor core neutronic and thermal hydraulic behaviour. Fuel behaviour and fission product transport models will be developed within the fuels program and graphite performance models within the materials program.

III.C The People's Republic of China

The Chinese high-temperature gas reactor R&D program is directed toward the development of the prototype modular reactor and conversion of the HTR-10 to a vertical shaft direct Brayton cycle electricity generator. The modular reactor R&D includes fuel fabrication, optimizing the Rankine cycle, and improved constructability. Much of the technology used in the modular reactor is based upon proven fossil power conversion technology. The HTR-10 modification replaces the existing steam generator Rankine cycle with a vertical Brayton cycle power conversion unit using active magnetic bearings. The R&D includes the development of test loops for the helium compressor and active magnetic bearings. Managing the damping, critical speeds, and system stiffness are critical elements in the successful deployment of such a vertical shaft machine. These issues will be evaluated in the R&D program.

The prototype pebble bed reactor (HTR-PM) is a modular 200 MWe reactor. The HTR-PM project received environmental clearance in March 2008 for construction start in 2009 and commissioning by 2013. Additional gas reactor modules are proposed for the HTR-PM site in the Shandong Province.

III.D. The Republic of South Africa

The South African R&D program is structured to support the deployment of the PBMR for either electricity generation purposes or process heat.

The R&D includes fuel manufacturing development, irradiation and testing, testing of key active components in their Helium Test Facility, testing of heat transfer mechanisms, and thermal hydraulic phenomena in their Heat Transfer Test Facilities (high temperature and high pressure). In the safety assurance area, R&D tasks include the effects of corrosion due to air ingress on the natural circulation potential of the PBMR primary circuit. Experiments with the critical facility and the PBMR micro-model have assisted in the development of the overall safety case through benchmarking of the analysis software.

PBMR is evaluating a move in product emphasis from a direct Brayton cycle to a steam co-generation Rankine cycle to match nearer term applications in process heat. This switch includes lowering the reactor outlet temperature to approximately 750°C and smaller power output around 200 MWt. This strategy is being made with consultation of potential customers in both South Africa and the United States. It also brings the South African development effort in better alignment with the Chinese HTR Program. The decision should be finalized in June 2009 after it is reviewed with the PBMR Board of Directors and the South Africa government, assuming the business case for this product design is positive.

IV. FUTURE USE OF GAS REACTORS

In all reasonable forecasts, nuclear energy must play an ever increasing role in the generation of electricity, which accounts for roughly one-third of the global, man-made CO_2 emissions. However, if we are to deal effectively with the combined threat of climate change and energy security, nuclear energy must expand its role beyond the generation of electricity. In the form of HTRs and VHTRs, nuclear energy can provide CO_2 emission-free process heat for chemical plants, refineries, and for the development of unconventional, hydrocarbon resources. In this context, a gas reactor is helium cooled, graphite moderated reactor with core outlet temperatures equal to or in excess of 750°C. Eventually, gas reactors will have sufficiently high core outlet temperatures to produce hydrogen from water to provide transportation fuels and serve as an excellent energy carrier, similar to the role of high-voltage transmission systems. Similarly, hydrogen permits the production of gases and liquids from the world's most abundant unconventional hydrocarbon, coal, without the emission of large quantities of CO_2.

These are not new ideas. In a 1982 overview paper, [1] the incentives for developing and deploying high-temperature gas-cooled reactors include "the efficient application of nuclear heat to:

1. Replace fossil fuels in a substantial portion of industrial process heat energy needs.
2. Significantly reduce the potential for the adverse environmental impact of CO_2 and other pollutants from burning fossil fuels for the production of process steam and heat.
3. Directly substitute for the heat formed by burning fossil feedstock fuels in the production of synfuels, syngas, and hydrogen.
4. Provide a path to nearly inexhaustible hydrogen fuel economy through thermo-chemical water splitting.

More recently, Konefal [2] produced an excellent overview of the potential applications of gas reactors for process energy applications. The report is a thorough, state-of-the-art assessment of the theoretical deployment of gas reactors for use in the following applications.

- Petroleum refining.
- Oil recovery.
- Coal and natural gas derivatives.
- Petrochemicals.
- Industrial gases, particularly hydrogen.
- Ammonia and nitrates production for inorganic fertilizers.
- Metals.
- Polymers.
- Cement.
- Pharmaceuticals.
- Paper and Glass.

Of the applications, the first seven are considered to be high priority. The application of nuclear energy to the generation of electricity, process heat, nonconventional hydrocarbon, and hydrogen are briefly discussed in the following sections.

Steve Aumier [3] of the Idaho National Laboratory proposes a vision of hybrid nuclear/conventional energy as a means to address the key aspects of energy security and climate change. Figure 1 illustrates some key elements of this vision.

IV.A. Electricity Generation

Electricity is the one area where nuclear energy is currently used commercially. Traditionally, commercial nuclear power plants are among the largest of the central generating stations. The Advanced LWRs are under construction for baseload electricity generation

Figure 1: Potential Application of High Temperature Gas cooled Reactors for a Broad Range of Hybrid Energy Applications

with capacity greater than 1 200 MWe. These proven, safe and reliable units will be the nuclear generation option of choice for most of the world's large utilities. However, other utilities prefer to build plants of a smaller scale, in increments of 250 MWe, as seen in the large number of natural gas-fired plants deployed in the 1990s throughout the world. If smaller nuclear plants can be licensed and built economically, the displacement of natural gas-fired combined cycle (NGCC) generation plants is likely to be significant in the future. This displacement will occur if the price of natural gas remains high and the price of CO_2 becomes significant. Proposed gas reactors are of a similar size as the current NGCC plants. In this vein, the smaller gas reactors can fill one or more important niches. These niches include either a situation where an electric utility does not need 1 200 MWe or more of generation or one where a large investment of 4B provides too great an exposure for the corporation. Another potential niche is for regions of the world where water usage is a critical issue for power generation, such as in the U.S., west and southwest. A gas reactor is small enough and efficient enough to use air-cooled condensers without a severe heat rate penalty. Air condensers are currently coupled to generating units with capacities as high as 500 MWe. The water situation will only exacerbate the energy crisis with time.

IV.B. Process Heat Applications

The second area of applications for future gas reactors is in the refining of crude and other feedstocks into transportation fuels. Worldwide, refineries produce 13% of all of the manmade CO_2 and consume large amounts of energy. For example, refineries use about 7.5% of the entire U.S. energy supply. The potential substitution of nuclear produced process heat for process heat produced by the combustion of high-quality fossil fuels, such as natural gas is potentially significant, if the economics are sensible.

The potential use of nuclear energy to displace the use of natural gas in an oil refinery and to reduce CO_2 emissions can be assessed by investigating the types of fuels and energy usage in a typical U.S. refinery, see Figure 2.

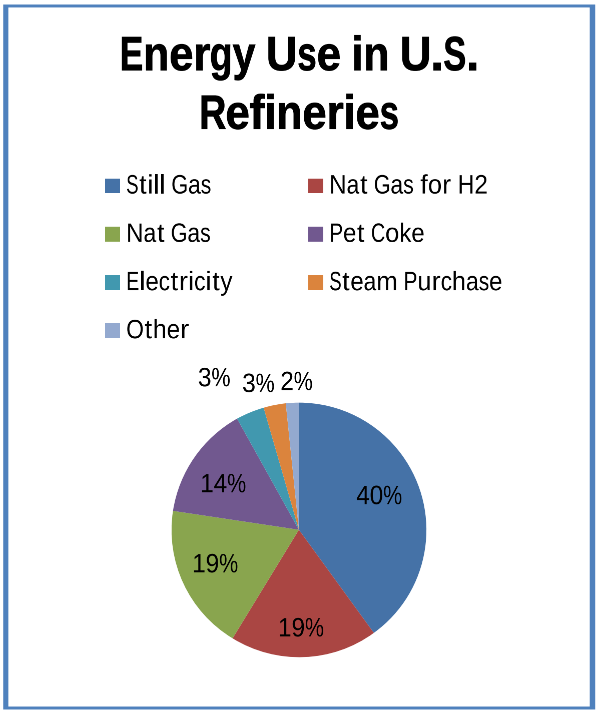

Figure 2: Sources of energy for the refining of crude oil in the average U.S. refinery

Discussions with oil companies suggest that nuclear energy could replace some but not all of the energy requirements of an oil refinery. At a minimum, 25% of the energy, the natural gas, the purchased steam and electricity can be replaced with nuclear energy in the largest refineries. In the United States alone, refineries use over 2.70 trillion cubic feet (TCF) of natural gas per year. Reference [8] identifies the thermal load (steam, heat and electricity) for a 200 000 bbl/d refinery as about 1 100 MWt. It is expected that economically replaceable energy is at most one-half the total, based upon these end-user discussions.

The chemical industry is another industry in which gas reactors could play an important role in the future. The three most energy consumptive chemical processes are the production of (1) ethylene, (2) ammonia and (3) chlorine. Ammonia, a key component of inorganic fertilizer production, requires copious amounts of natural gas in its production. In a chemical plant, natural gas is used for feedstock and process heat to provide steam and electricity through co-generation. To quantify the energy use and CO_2 emitted in the manufacture of chemicals, key data for the U.S. are:

- The chemical industry uses about 15% of the energy used in the U.S.

- The production of ethylene, a major petrochemical product, requires about 4.9 QUAD of energy annually and represents 34% of the entire petrochemical annual energy usage.

- The total natural gas used annually in the production of ammonia is 6 TCF, with 42% burned to produce the heat necessary for ammonia production.

- Ammonia production creates about 115 million MTs/y of CO_2. Annual chlorine production requires 5 QUAD of energy, mainly in the form of electricity.

IV.C. Unconventional Hydrocarbon Production

In order to increase the energy security of any country in the world, that country must be in a position to utilize its indigenous resources. In the United States, for example, petroleum feedstock could be produced from domestic, unconventional hydro-carbon sources, such as heavy oil and tar sands, coal, biomass and oil shale. The United States has huge deposits of both coal and oil shale and could produce as much as 1.3 billion dry tons per year of sustainable biomass from agriculture and forest residues. [4]

Chevron [4] recently completed a thorough evaluation of the potential uses of nuclear energy in the exploitation of unconventional hydro-carbons in their future business. In their evaluation, the unconventional hydrocarbons include heavy oils using steam floods, oil sands using cyclic steam stimulation and steam-assisted gravity drainage, oil shale, and coal to liquids.

IV.C.1. Heavy Oil and Tar Sands

The recovery of heavy oil and the bitumen recovery from tar sands require significant amounts of steam and electricity for heating and pumping the product. The top four resources of tar sands in the world are: Canada, Venezuela, Columbia and Russia. For example in North America, if we produce 800 000 bbls/day from a combination of 50% heavy oil and 50% tar sands, Bradruzzaman [5] concludes that about 0.314 TCF of natural gas per year is required with an attendant CO_2 release of over 21 million MT per annum. The natural gas cost at today's price of $6 million BTU delivered is over $5 million per day. With a target of 0.8 MMbbls/d from heavy oil and tar sands, between 15 and 25 gas reactors are required, assuming that a 500 MWt can provide sufficient energy to produce 50 Mbbls/d. The number of reactors deployed depends upon the layout of the fields.

IV.C.2. Coal to Liquids and Gases

Coal is one of the most abundant hydro-carbons on earth. The top coal resources in the world are located in Russia, the United States, China, Australia, and Canada. While coal is currently used to produce electricity, it can also serve as transportation fuel. The process for converting coal to liquids (CTL) was developed in Germany in the 1920s and by World War II became the source of 90% of that nation's liquid fuel requirements. Nine indirect and 18 direct liquefaction plants produced four million metric tonnes per year. Later, as a result of the apartheid based embargoes, South Africa, using technology similar to that used by the Germans, developed their own CTL industry that now produces up to 10 million metric tonnes per year meeting about 40% of the country's current liquid fuel needs. There is also a growing interest in other countries with major coal reserves, *e.g.* the United States and China, to develop processes that can exploit the large coal deposits to meet their growing petroleum requirements. For example, if the coal deposits in the United States were converted to liquid hydrocarbons, they would represent over 60% of the world's proven oil reserves. China is experiencing growth in coal liquefaction as a way of utilizing its coal reserves and reducing its dependence on imported oil. The South African company, Sasol, is planning two CTL plants in China. [6] In the United States, some nine states are actively considering CTL plants. Global liquid coal production is expected to rise from 150 000 bpd today to 600 000 in 2020 and 1.8 million bpd in 2030. [7] Currently, the CTL plants produce over 32 million MT of CO_2/year. The plants are the largest, global, single point sources of CO_2 emissions. Expanded deployment of conventional CTL plants is a major environ–mental concern.

A modified FT CTL plant [8] can be designed and built that produces very little CO_2. The large amount of CO_2 produced in conventional FT-CTL plants is one of the primary objections to the use of coal to make refining feedstock. In the modified FT-CTL plant, gas reactors are used to split water to produce the oxygen (O_2) and hydrogen (H_2), in lieu of the air separation unit and the inherent water shift reaction. The modified CTL plant uses 40% less coal to produce the same amount of product. The oxygen derived from the water-splitting process is used for coal gasification, thus eliminating the need for air separation units, which represent about 10% of the total cost of the plant. However, the modified FT-CTL process requires a large amount of H_2, on the order of 0.22 kg of H_2 for each kg of coal.

IV.C.3. Oil Shale

The oil shale deposits in the world contain the oil equivalent of over twice the proven crude oil reserves. The United States has 60% of oil shale in the world, representing 7 trillion barrels of oil equivalent. Estonia, Australia, China, and Brazil have the next largest oil shale resources. It is estimated that the oil shale deposits from the Green River region of the western United States will yield over 1 million barrels per acre. While the in-situ extraction of kerogen, the useful product, from oil shale, is in the experimental stage at Shell [9], scientists estimate that it requires approximately 12 GWt to produce 1 MMbbls/d. [10] The current approach is to use electric down hole heaters to raise the temperature of the oil shale to a level sufficient (~370°C) to release the kerogen. Over 250 KW-hr of electricity is required to produce one barrel of kerogen. With the current generation mix in the United States, the annual CO_2 emitted in the generation of the electricity required for the in-situ oil shale production of 0.4 MMbbls/d is about 17.5 million MT. An alternate, more energy efficient approach is the use of hot fluids to heat the ground rather than incurring the conversion losses in generating electricity. One nonemitting source of heat for the in-situ oil shale is nuclear in the form of gas reactors. If gas reactors produced the heat for the *in-situ* oil shale production of 0.4 MMbbls/d, between 10 and 15 gas reactors (500 MWt) are required, depending upon the morphology of the oil shale deposits.

IV.C.4. Biomass

The conversion of biomass to liquids is more confounded than the other unconventional hydrocarbon sources, because so many different ways to produce liquids exists. Forsberg [11] provides an excellent description of the different processes and biomass feedstocks available or under development. Biomass can be converted to liquids using the Fischer-Tropsch process, similar to the conversion of coal to liquids. However, in the case of biomass, the net CO_2 burden is substantially less than for coal. This assumes that an equivalent amount of CO_2 produced in the gasification process is consumed in the growing of the crops, a.k.a., terrestrial sequestration. Gas reactors could produce the steam and the electricity required for the biomass conversion plants, reducing the CO_2 emitted from the burning of fossil fuels.

IV.D. Hydrogen Production

There is a growing need for hydrogen, emission-free process heat and steam for industries and process heat and steam for the production of unconventional hydrocarbons, as shown in previous sections.

Hydrogen is required in ever-growing quantities to process the lower quality, higher sulfur crude oil that is available today. The current method for producing hydrogen is through steam methane reformation of natural gas. The increasing price volatility of natural gas and the strong potential for carbon constraints are reasons for developing alternative means for producing hydrogen. Nuclear energy can produce emission-free hydrogen in a number of ways, including:

- conventional water electrolysis (using nuclear generated electricity),

- high-temperature electrolysis (using nuclear generated electricity and steam) requires temperatures up to 900°C for 50% conversion efficiency,
- thermo-chemical cycles water splitting (using nuclear heat) requires temperatures of 850°C,
- hybrid cycles (combining thermo-chemical and electrolytic steps) also requires temperatures of 850°C, and
- steam methane reforming (SMR) (using nuclear energy for the endothermic heat of reaction and steam), requires temperatures of 800°C.

The economics of hydrogen production through water splitting, the price of natural gas and the price of CO_2 emissions will determine the actual deployment of gas reactors for hydrogen production.

V. CONCLUSION

This gas reactor review concludes the following:

1. While not all of the gas-cooled reactors demonstrations and deployments satisfied every expectation, all of them did provide valuable information and experience.

   Lessons learned include:

   a. Lower temperature CO_2 cooled, natural uranium fuelled reactors can be successfully deployed for periods in excess of original design life.

   b. Helium can be successfully used as a coolant and graphite as a moderator for temperatures up to 950°C.

   c. Coated fuel particles, particularly of the TRISO type, perform reliably to high burn up levels, when properly manufactured.

   d. Conventional LWR type containments are not required for gas-cooled reactors.

   e. Fuel handling and refuelling of both prismatic blocks and pebbles can be reliably performed.

   f. Control rods can be confidently inserted into a bed of fuel pebbles.

   g. Tests demonstrate the safety of gas reactors under loss-of-coolant flow without scram conditions.

2. Current research programs within GIF and specific country programs address major development, demonstration and deployment issues. These include fuel, materials, constructability and manufacturability, turbo machinery and hydrogen production.

3. Gas reactors can reduce CO_2 emissions and increase energy security through their ability to:

   a. Play a niche role in electricity generation, where larger advanced light water reactors are either too large, too capital intensive or too water consumptive.

   b. Provide industrial process heat for oil refining, chemical, petro-chemical and fertilizer production with a minimum generation of CO_2 emissions and maximum conservation of strategic hydro-carbon resources, particularly natural gas.

   c. Support the development of unconventional hydrocarbons to increase national energy security. These unconventional hydro-carbons include heavy oil and tar sands, coal (in conjunction with hydrogen production), oil shale, and biomass.

   d. Provide the thermal energy for hydrogen production using thermal chemical, high-temperature electrolysis or hybrid processes.

Nomenclature

AGR – Advanced Gas Reactor
ASME – American Society of Mechanical Engineers
ASTM – American Society for Testing and Materials
AVR – Arbeitsgemeinschaft Versuch Reaktor
bbls/d – barrels (42 gallons)/day
BISO – Buffered isotropic pyrolytic carbon fuel particle
BTU – British Thermal Unit
C-C – Carbon-Carbon
CEGB – Central Electricity Generating Board
CTL– Coal To Liquids
EdF – Electricite de France
FSV – Fort Saint Vrain
FT – Fischer-Tropsch process
GA – General Atomics company
GIF – Generation IV International Forum
GWe – Gigawatt (electric)
HTE – High Temperature Electrolysis
HTR – High Temperature Reactor
HTR 10 – High Temperature Reactor – 10 MWe
HTR PM – High Temperature Reactor – Prototype Modular
HTTR – High Temperature Test Reactor
M – Thousand (common terminology in petroleum industry)
MM – Million (common terminology in petroleum industry)
MTs/y – Metric Tonnes per year
MWe – Megawatt (electric)
MWt – Megawatt (thermal)
NGCC – Natural Gas fired Combined Cycle
NGNP – Next Generation Nuclear Project
OECD – Organization for Economic Co-operation and Development
PBMR – Pebble Bed Modular Reactor
QUAD – Quadrillion BTU
RCCS – Reactor Cavity Cooling System
RPV – Reactor Pressure Vessel
SiC-SiC – Silicon Carbide-Silicon Carbide
SMR – Steam Methane Reforming
TCF – Trillion Cubic Feet
THTR – The Thorium High Temperature Reactor
TRISO – Tri-structural-isotropic fuel
UKAEA – United Kingdom Atomic Energy Agency
USAEC – United States Atomic Energy Commission
VHTR – Very High Temperature Reactor

References

1. Moore, R.A., *et al.*, “HTGR experience, programs and future applications”, Nuclear Engineering and Design 72, North Holland Publishing Company, 1982, pp. 153-174
2. Konefal, J., “Survey of HTGR Process Energy Applications”, MPR-3181, May 2008 Draft Report for INL

3. Steven Aumier, "Hybrid Energy Systems – Nuclear Energy for the Twenty-First Century", a presentation to the Department of Energy – Nuclear Energy, March 2009, Washington, DC.

4. Marano, J.J. *et al*., "Life-cycle Greenhouse Gas Emissions Inventory for Fischer-Tropsch fuels", prepared for the US DOE National Energy Technology Laboratory, 2001

5. Bradruzzaman, A., *et al*., "Nuclear Energy for Unconventional Fossil-fuel Resource Recovery", SPE 116294, 2008 SPE Annual Technical Conference, Society of Petroleum Engineers, Sept. 2008

6. "Sasol Plans Two Coal-to-liquid Fuel Projects", published in China Daily on January 30, 2007; downloaded at www.china.org.cn/english/BAT/198162.htm

7. Newsweek "Special Energy Edition", Dec 2006- Feb 2007

8. Cherry, R.S. *et al*., "Use of Nuclear High Temperature Reactor in a Coal-to-Liquids Process", INL Report Ext 06-11667, Aug 2006

9. John Birger, "Oil from a stone", Fortune Magazine, November 1, 2007

10. Forsberg, C., "Use of High-Temperature Reactor Heat in Refineries, Underground Refining, and Bio-refineries for Liquid Fuels Production", HTR 2008-58226, Proceedings of the 4th Annual International Topical Meeting on High Temperature Reactor Technology, Sept. 2008, Washington, DC

11. Forsberg, C., "Meeting U.S. Liquid Transport Fuel Needs with a Nuclear Hydrogen Biomass System", International Journal of Hydrogen Energy (in Press), 2008

VHTR – ONGOING INTERNATIONAL PROJECTS

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I. INTRODUCTION

The world today is facing tremendous energy challenges as a result of both a demographic explosion worldwide and a fast economic development of China and India that represent 30% of the world's population. Most conservative scenarios will drive the energy demand to high levels when at the same time fossil resources are becoming scarcer and should be replaced by low carbon energy sources to limit CO_2 emissions and associated risks of climate change. Meeting the fast growing energy demand while decreasing greenhouse gas emissions calls for making an extensive use of renewable and nuclear energies to displace fossil fuels for producing electricity and other energy products such as fuels for air- and ground-transportation, as well as process heat for the industry (petro-chemistry, steel making and others…). Indeed, electricity is responsible for ~30% only of CO_2 emissions worldwide. Light water reactors can generate nuclear electricity and hydrogen through alkaline electrolysis. However, the unique capability of *Very High Temperature Reactors (VHTRs)* to produce process heat above 600°C makes them a strategic reactor type that can more efficiently produce hydrogen through steam electrolysis, or supply both hydrogen and high temperature heat for producing synthetic fuels from coal or biomass, or also supply high temperature heat and hydrogen or syngas as chemical reactant to varied industrial plants including petro-chemistry and steelmaking. Based on the past experience acquired from the 1960s through the 1980s on experimental high temperature reactors (HTRs) and prototypes, new incentives for non electricity nuclear productions add up to the attractive safety features of medium size HTRs (< 600 MW_{th}) to make VHTR the system that fosters today the most active R&D cooperation in the frame of the Generation IV International Forum (GIF) and the greatest number of national projects of prototypes in the next two decades.

II. PAST EXPERIENCE ON HTR

In the 1960s two different types of reactors were designed and built, primarily to produce electricity. Experimental HTRs with a prismatic block-core were developed in United Kingdom *(DRAGON reactor, 20 MW_{th})* and the United States *(Peach Bottom, 40 MW_e)*. They were followed by the prototype of *Fort St. Vrain Generating Station (330 MW_e)* that operated from 1979 to 1989. This reactor established the technical feasibility of HTRs even though it was beset by problems of power fluctuations, jamming of control rod and leakage of water into the core which finally caused its decommissioning for economic reasons.

Over the same period, Germany developed pebble bed reactors and built an experimental reactor *(AVR, 15 MW_e)* on the Research Centre of Jülich that successfully operated from 1966 to 1987 and gave valuable feedback on pebble fuel and overall operation. Following this experience, a 300 MW_e prototype of power reactor that was aimed at using thorium fuel was built and operated: *the Thorium High Temperature Reactor (THTR-300, 300 MW_e)*. This prototype however suffered a number of technical difficulties and was finally closed in 0icity production. No further developments were to occur until the late 1990s when the interest in HTRs was revived by needs of low carbon high temperature heat supply for varied industrial processes.

III. TODAY'S CONTEXT

First, the Japan Atomic Energy Agency (JAEA) built a research reactor in Oarai, the High Temperature engineering Test Reactor (HTTR) that was put in service in 1998 and reached its full design power of 30 MW_{th} in 1999 with an outlet helium temperature of 850°C. Subsequent tests have demonstrated the safe behavior of the reactor in various accidental sequences and the successful operation at the design temperature of 950°C. The HTTR is to restart in 2009 after 18 months at shutdown, and to proceed with a continuous operation at 950°C for 60 days. In parallel with tests on the HTTR, JAEA is developing the sulfur-iodine thermo-chemical process to produce hydrogen. A first demonstration of this process was achieved in 2003 when a continuous production of 30 litres of hydrogen per hour was obtained for a few days. The next steps are tests of a pilot plant of 400 kW (30 m^3/hr) around 2012 and tests of nuclear production coupled to the HTTR at pre-industrial scale (10 MW and 1 000 m^3/hr) around 2015-2020.

Then, Institute of Nuclear and New Energy Technology (INET) of Tsinghua University in China built the experimental reactor HTR-10 (10 MW_{th}) that was put in operation in 2000. The successful operation of this reactor demonstrated the updated pebble bed core HTR technology and paved the way for scaling up this technology into the HTR-PM project in China.

Currently, the revival of interest in high temperature process heat applications fostered R&D and projects of new builds of HTRs in the world thus preparing the advent of a new generation of this reactor type: the Very High Temperature Reactor (VHTR).

IV. ON GOING INTERNATIONAL PROJECTS

There are today in the world several projects of VHTR prototypes planned for the period 2015-2025. They are at different stages of maturity and aim at varied applications: electricity first and process heat in a second stage, or dedication to hydrogen production. The interest and support of end user industries is sought to create private / public partnerships to build and operate such prototypes and proceed with demonstrations relevant to their industrial needs. Industrial sectors concerned include the oil industry (extraction & treatment of oil sands, production of synthetic fuels from coal & biomass), as well as chemical and steel industries.

IV-A – HTR-PM in China

In 2005, China announced its intention to scale up the HTR-10 technology and to realise a national project of 200 MW_e MHTGR commercial plant with independent intellectual property rights. This project consists in two High Temperature Reactor-Pebble Bed Modules (HTR-PM)[1] of 250 MW_{th} with a helium core outlet temperature of 750°C that drive together a steam turbine of 200 MW_e. It is supported by a 3-party joint venture: the industry, the university and research organizations. The main design features of the nuclear island that were selected in 2006 are largely derived from those of the HTR-10. The basic design is completed and the preliminary safety analysis report is under review.

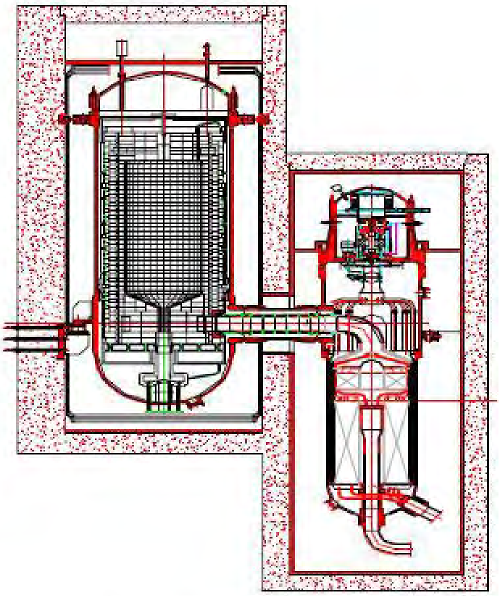

Figure 1: Primary system of HTR-PM

The construction has begun in 2009 on the site of the Shidaowan plant in the Province of Shandong with a commissioning planned in 2013. As first demonstration power plant, the HTR-PM, is not expected to be an economically self-financing project and hence the government partly funds its construction and operation so as to support the operation of the plant and guarantee the owner a fair recovery of its investment. The economic competitiveness of HTR-PM power plants is sought through modularization, batch construction and use of mature technologies to the extent possible to reduce technical risks. In this respect, best use will be made of the successful experience gained from the HTR-10 and other HTR projects abroad.

The lay-out of the nuclear island and overall design features of the HTR-PM are similar to those of the HTR-10, which have been tested for several years of operation. The conventional island will use the mature technology of high temperature and high pressure over-heat steam turbine-generator which is widely used in thermal power plants. The manufacture of fuel elements will also be based on the technology verified on the HTR-10 project. The key systems and equipments will be tested on engineering scale experimental rigs in order to guarantee the safety and reliability of the HTR-PM project. In addition, the use of mature technologies and successful experiences developed abroad is also considered through international cooperation.

After the HTR-PM demonstration plant has demonstrated a successful operation, larger scale HTR-PM power plants using multiple-modules and one steam turbine-generator will be built so as to take full benefit from standardization and modularization permitted by the technology.

A comprehensive plan supported by the government has been defined to develop and test key technologies and specific engineering features for the HTR-PM. A HTR-PM engineering laboratory and a large helium engineering testing loop, as well other large scale testing rigs are under construction at INET to test the main components of the reactor. At the same time, a fuel production line with a capacity of 300 000 fuel pebbles per year will be built in Inner Mongolia to serve HTR-PM projects.

Even though aimed operating conditions in a first stage correspond to a core outlet temperature of 750°C, the reactor is designed to achieve a core outlet temperature of 950°C with current core design and fuel element technologies. Improvements of fuel performances should enable to reach ultimately a core exit temperature of 1 000°C. Besides, the modular nature of the HTR-PM makes it possible to replace the steam turbine of the power conversion system by a helium turbine or a super critical steam turbine, as well as by a hydrogen production plant in a second stage.

IV-B – Pebble Bed Modular Reactor (PBMR) in the Republic of South Africa

Pebble Bed Modular Reactor Pty. Ltd (PBMR)[2] is a public-private partnership that was established in 1999 in the Republic of South Africa to initiate the development of a modular pebble-bed reactor with a rated capacity of 165 MW_e. This design featured a thermal power of 400 MW_{th} and a direct power conversion with a gas turbine operating with an inlet helium temperature of 750-900°C. In June 2003 the government of the Republic of South Africa approved a prototype of pebble-bed modular reactor of 110 MW_e for Eskom on the site of Koeberg. This prototype that was intended to be put in service in 2014 was meant to precede a series of 24 PBMRs so as to make up 4 000 MW_e out of the 12 000 MW_e additional nuclear capacity planned by 2030. Facilities dedicated to PBMR specific technologies testing have been realized in 2007: a "Heat Transfer Test Facility", a "Helium Test Facility", a "Pebble Bed Micro Model" and an "Electro-magnetic blower". A fuel laboratory developed manufacturing processes of TRISO fuel particles and quality assurance testing techniques in collaboration with NECSA and successfully manufactured coated fuel particles with enriched uranium in December 2008.

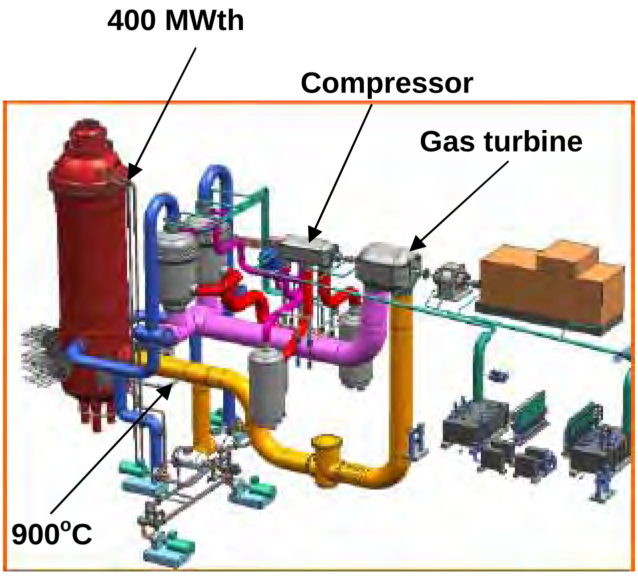

Figure 2: Lay-out of initial 165 MW_e PBMR project

In 2009 the PBMR project, like other projects of nuclear equipment in South Africa, faced funding difficulties and happened to have its business plan re-oriented towards the supply of industrial process heat. The current focus of the PBMR is on onsite power, cogeneration, desalination and direct process heat delivery. Target process heat applications include coal to liquid or gaseous fuels, petrochemicals, ammonia/fertilizer, refineries, oil sand recovery, bulk hydrogen for future transportation and water desalination. Thus, PBMR Ltd started developing options for commercial fleets with Sasol for producing synthetic fuels from coal, with Eskom for electricity, as well as with US and Canadian cogeneration end users including oil sand producers. The PBMR project was accordingly revisited to develop one standard design that meets all requirements for these varied applications, thus leading to a cogeneration steam plant with a thermal power of 200 MW_{th}, a helium temperature of 750°C at core outlet and a steam generator directly placed in the primary loop. A conventional sub-critical steam turbine is selected for first generation plants whereas super-critical cycles can be considered for next generation plants.

IV-C – Next Generation Nuclear Project (NGNP) in the United States

US-DOE initiated exploration of the NGNP[3] concept as part of the Generation IV Nuclear Systems Initiative in 2003. The NGNP project was then mandated by the US Energy Policy Act of August 8, 2005 as a high-temperature gas-cooled reactor intended for high-efficiency electricity production, high-temperature process heat generation, and nuclear-assisted hydrogen production at the Idaho National Laboratory (INL). It would be co-located with an industrial plant that would use process heat from the reactor and could operate in 2021. Pre-conceptual and conceptual design studies have been conducted under contracts awarded in 2006 and 2008 by US-DOE to AREVA, General Atomics and Westinghouse. General Atomics and AREVA are putting forward their GT-MHR and Antares concepts of prismatic block-type high temperature reactor whereas Westinghouse is supporting the Pebble Bed Modular Reactor. The current NGNP concept employs an indirect power conversion that uses intermediate heat exchangers to transfer heat from the reactor primary loop. The secondary loop may be used as a heat source for the production of electricity, hydrogen, or other industrial uses. A number of studies as a part of the conceptual design have identified bounding conditions as follows: i) At this time there are no discriminating technical factors that favor pebble bed or prismatic design over another, ii) One-size-fits-all approach is not necessarily consistent with all off the end user needs, and iii) User needs indicate that the initial gas outlet temperature will be in the 750-800°C range. However, R&D will continue to enable full potential as well (950°C).

Fuel development irradiation is being conducted at the INL Advanced Test Reactor on high temperature fuel kernels made at BWXT and coated at Oak Ridge National Laboratory. Additional research is also proceeding to better understand irradiated graphite stability under load at operating temperatures, qualify high-temperature metallic alloys, and to support development of physics, thermo-fluids, and accident simulation codes. The NGNP project took another step in August 2008 when the US-DOE and the NRC submitted a joint licensing plan leading to a licence application filed in 2013. DOE is currently developing a final strategy for partnering with the industry (nuclear vendors and potential users of process heat in sectors such as oil-, chemistry or steelmaking) to drive the development of the NGNP project.

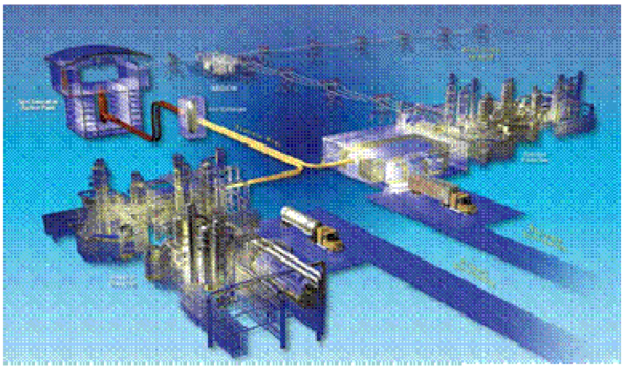

Figure 3: Artist view of NGNP supporting both applications of nuclear production of hydrogen and synthetic hydrocarbon fuel from coal

IV-D – Gas Turbine High Temperature Reactor (GTHTR-300C) in Japan

The Japan Atomic Energy Agency (JAEA) is currently conducting research and development for the project of "*Gas Turbine High Temperature Reactor 300–Cogeneration*" (*GTHTR300C*)[4] (Figure 4) that is dedicated to CO_2 emission free cogeneration of electricity and hydrogen by sulfur-iodine thermo-chemical water splitting process.

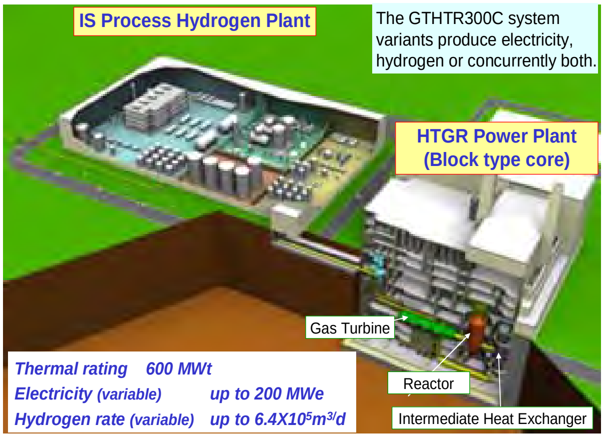

Figure 4: Gas Turbine High Temperature Reactor 300 for Cogeneration (GTHTR300C)

With a thermal power of 600 MW and a block-type core with an exit temperature of 950°C, the GTHTR300C is believed to be highly efficient and economically competitive for cogenerated hydrogen and electricity. The Intermediate Heat Exchanger (IHX) and the gas turbine are installed in series in the primary circuit so that heat over 900°C (170 MW_{th}) can be efficiently used for hydrogen production and helium at 850°C can be used for generating electricity. The GTHTR300C thus allows a co-generation of about 200 MW_e and 640 000 m^3/day of hydrogen by the thermo-chemical sulfur-iodine process (enough to serve about 170 000 fuel cell vehicles).

In order to minimize cost and risk of deployment, the GTHTR300C is based on HTTR-derivative technologies, on current helium turbine power conversion and on technologies under development for the thermo-chemical water splitting process. A technology roadmap of nuclear hydrogen production was issued by the Atomic Energy Commission of Japan in July 2008. It envisions the introduction of commercial HTGR hydrogen production around 2030 and foresees by 2020 a prototype of commercial reactor based on technology and reliability demonstrations achievable in the HTTR and the associated system dedicated to pre-industrial sulfur-iodine cycle demonstrations.

IV-E – Nuclear Hydrogen Development and Demonstration (NHDD) in Korea

In a context of wilful development of hydrogen technologies to prepare the hydrogen economy in the Republic of Korea, the Korean Atomic Energy Commission approved a national nuclear hydrogen program in Dec. ’08 that consists of two major projects:

- A project of key technologies development for nuclear hydrogen, and
- A project of Nuclear Hydrogen Development and Demonstration (NHDD).[5]

The project of key technologies development was launched at KAERI in 2006. It focuses on the development and validation of technologies that are key to nuclear hydrogen systems. Topics involved include design and computational tools, high-temperature materials and components, TRISO fuel particle manufacturing and performance testing, and the sulfur-iodine thermo-chemical hydrogen production process. The project will extend up to 2017 in phase with goals of GIF’s and NHDD’s projects. The NHDD project aims at designing, constructing a nuclear hydrogen production system and demonstrating its safe and reliable operation. The project is expected to be launched in 2010 with target dates of 2022 for the completion of construction and 2026 for prototypical demonstrations. Reference options for such a nuclear hydrogen production system consist of a very high temperature reactor of 200 MW_{th} with a core outlet temperature of 950°C, 5 modules of hydrogen production based on the sulfur-iodine water-splitting process, and an intermediate heat transport loop between the reactor and the hydrogen plant. A cooled reactor vessel design is adopted to make use of domestic manufacturing capabilities. Both the prismatic block and the pebble bed cores are considered at this stage. Commercial prospects are at an early stage of discussion.

IV-F – Multinational cooperation on the VHTR System in the Generation IV International Forum[6]

The potential of a VHTR at 900-1 000°C to match temperature requirements for advanced

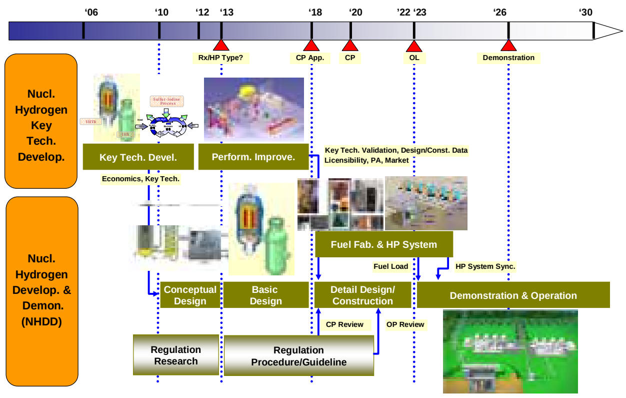

Figure 5: National Nuclear Hydrogen Project Plan (approved by the AEC of Republic of Korea in Dec. ‘08)

hydrogen production processes based on electro- or thermo-chemical water splitting processes was the initial driver for this reactor type to be selected in 2002 among the six Generation IV Systems. Missions of the VHTR have expanded since then to cogeneration of electricity and process heat for varied industrial applications. This system experiences a sustained interest from all active members of the GIF since its beginning. The VHTR System Arrangement was signed in December 2006 by Canada, EURATOM, France, Japan, the Republic of Korea, Switzerland and the United-States. The People’s Republic of China signed this Arrangement in October 2008 and the Republic of South Africa is expected to sign it in 2009. Multinational cooperation in the GIF complements national R&D efforts for current projects of reactor at 700-850°C and also develops technology breakthroughs for the VHTR aiming at 900-1 000°C. Projects on “Fuel and fuel cycle” and “Hydrogen production” became effective in January and March 2008 and a project on “Materials” will become effective in the fall of 2009. A project on “Computational methods, validation and bench–marking” will be ready for signature at the end of 2009. Cooperative work on TRISO fuel includes sharing irradiation experiments, post irradiation evaluation facilities and constituent materials properties. Cooperation on hydrogen production processes allowed to share the realization and results of laboratory scale experiments on the sulfur-iodine and high temperature electrolysis, to advance the development of catalysts and share results of technical and economic assessments of varied candidate water splitting processes. Cooperative development of materials covers graphite, advanced super-alloys (nickel-based and

9Cr ferritic steels) and composite ceramics. Results are compiled in a common data base operated by the Oak Ridge National Laboratory. Specific Agreements will be worked out to frame exchanges between cooperative R&D in the GIF and VHTR related projects so as to assure a fair treatment of R&D results generated by GIF members and their privileged access to operating parameters of prototype reactors in fair conditions.

IV-G – HTR Technology Network and ooperative R&D in Europe: towards a Demonstrator?

A partnership of European nuclear industrial and research organisations has been established with the creation in 2000 of the (European) "HTR Technology Network" (HTR-TN) for developing HTR technology. HTR-TN has played since then a prominent role in defining a strategy for European R&D on HTRs and implementing this strategy in Euratom Framework Programmes (FP) since 2000 (5^{th} FP). This led to revive in the 6^{th} FP (2002-06) the past experience in Europe on HTR design tools and technologies (fuel, materials, helium systems' technology, coupling technologies…) in a program called RAPHAEL.[7] This set the stage for EURATOM to bring consistent contributions to VHTR R&D Projects in the Generation IV International Forum and for approaching industrial sectors potentially interested in low-carbon process heat. Investigating prospects of nuclear process heat applications for oil, chemical or steelmaking industries is currently in progress within the project Europairs that was launched in 2009 (FP7) and where potential end-users specify their needs and interact with the designers and safety authorities.

In order to achieve the industrial coupling between a nuclear heat source and industrial processes, the unfortunate scission between nuclear and non-nuclear communities and cooperative programs in Europe should be overcome. Besides, as the licensing of modular HTR/VHTRs and their coupling with chemical plants are critical issues, early interactions should be organised with regulators and Technical Safety Organizations. HTR/VHTR projects will continue in FP7, as initiated by RAPHAEL in FP6, and will contribute to VHTR R&D Projects of the Generation IV International Forum.

The launching in September 2007 of a Technology Platform on "Sustainable Nuclear Energy" (SNE-TP)[8] initiated a process of building an integrated and consistent program of R&D among European stakeholders along three directions: light water reactors, fast-neutron reactors with a closed fuel cycle and high temperature nuclear technologies for the cogeneration of non-electricity products.

Marketing prospects of high temperature nuclear heat are currently too uncertain for stakeholders of the nuclear industry and potential users of HTR energy products to envision yet building a prototype of next generation HTR in Europe. This issue will be debated versus the alternative that consists of having a significant European participation in a prototype abroad such as the NGNP in the US, PBMR in South Africa or HTR-PM in China.

V. FUTURE PROSPECTS

The unique capability of VHTRs to produce process heat above 600°C makes them an efficient reactor type to displace fossil fuels in a number of varied applications such as producing electricity, non-conventional hydrocarbon fuels from coal or biomass, and process heat for energy intensive industries (oil refining, petro-chemistry, chemistry, steelmaking…). Current research programs within GIF and specific country programs address major developments, demonstration and deployment issues. In particular, the multinational cooperation within the Generation IV International Forum allows to share efforts to advance VHTR technologies and to speed-up the development of breakthroughs for this reactor type. Furthermore, both experimental reactors in operation in Japan (HTTR) and in China (HTR-10) offer unique opportunities to qualify precursor VHTR technologies and design codes. Finally, ongoing projects of next generation HTR prototypes and projected pre-industrial demonstrations pave the way for the deployment worldwide of extended applications of nuclear power beyond the production of electricity and derived energy products that are accessible to Gen III light water reactors. These unique capabilities that enable increased reductions of CO_2 emissions, together with the versatility of VHTRs attest the high potential of this reactor type and spurs the interest of all GIF active members, as well as a growing participation in associated R&D as the GIF expands.

Acknowledgements

Authors acknowledge VHTR Steering Committee members for their inputs to this review paper of national and international VHTR related ongoing projects.

Nomenclature

AEC – Atomic Energy Commission
AVR – Arbeitsgemeinschaft Versuch Reaktor
FP6, FP7 – 6^{th}, 7^{th} European R&D Framework Programme
GIF – Generation IV International Forum
GT-MHR – Gas Turbine Modular Helium-cooled Reactor
HTR – High Temperature Reactor
HTR-10 – High Temperature Test Reactor (10 MW_{th})
HTR-PM – High Temperature Reactor – Pebble-bed Module
HTR-TN – High Temperature Reactor Technology Network
HTTR – High Temperature Test engineering Reactor
IHX – Intermediate Heat eXchanger
INET – Institute of Nuclear and New Energy Technology
INL – Idaho National Laboratory
KAERI – Korea Atomic Energy Research Institute
MHTGR – Modular High Temperature Gas-cooled Reactor
MW_e – Megawatt (electric)
MW_{th} – Megawatt (thermal)
NECSA – South African Nuclear Energy Corporation
NGNP – Next Generation Nuclear Project
NHDD – Nuclear Hydrogen Development and Demonstration
NRC – Nuclear Regulatory Commission
PBMR – Pebble Bed Modular Reactor
R&D – Research and Development
SNE-TP – European Sustainable Nuclear Energy Technology Platform
THTR – Thorium High Temperature Reactor
TRISO – Tri-Structural Isotropic Fuel
US-DOE – Department Of Energy of the United-States
VHTR – Very High Temperature Reactor

References

1. Zhang, Z., Z. Wu *et al*., "Current Status and Technical Description of Chinese 22 x 250 MW_{th} HTR-PM Demonstration Plant", Nuclear Engineering and Design, Vol. 239, Issue 7, July 2009
2. Kriek, J., "Status of the Pebble Bed Modular reactor Project", *Proc. of the 4th International Topical Meeting on High Temperature Reactor Technology HTR2008,* Washington, DC (USA) (2008)
3. Holbrook, M.R., and T. Cook, "NRC Licensing Strategy Development for the NGNP", *Proc. of the 4th International Topical Meeting on High Temperature Reactor Technology HTR2008,* Washington, DC (USA) (2008)
4. Kunitomi, K., *et al*., "JAEA's VHTR for Hydrogen and Electricity Cogeneration: GTHTR300C", Nuclear Engineering and Technology, Vol. 39 No.1 February 2007
5. Chang, J., "Status of Nuclear Hydrogen Production Technology Development Project in Korea", International Conference on Non-electrical Applications of Nuclear Power, Oarai (Japan) (2007)
6. Generation IV International Forum Annual Report (2008), OECD Nuclear Energy Agency, Issy-les-Moulineaux (2008)
7. RAPHAEL Journal No.4 September 2008.
8. Sustainable Nuclear Energy Technology Platform – Strategic Research Agenda 2009 (www.SNETP.eu)

THE VHTR FUEL AND FUEL CYCLE PROJECT: STATUS OF ONGOING RESEARCH AND RESULTS

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I. INTRODUCTION

The VHTR Fuel and Fuel Cycle project is intended to provide demonstrated solutions for the VHTR fuel (design, fabrication and qualification) and for its back-end management.

TRISO coated particles, which are the basic fuel concept for the VHTR, need to be qualified for relevant service conditions (Figure 1). Furthermore, its standard design UO2 kernel surrounded by successive layers of porous graphite, dense Pyrocarbon (PyC), silicon-carbide (SiC) then Pyrocarbon (PyC) could evolve along with the improvement of its performance through the use of UCO kernel or ZrC coating for enhanced burn-up capability, minimized fission product release and increased resistance to core heat-up accidents (above 1 600°C). Fuel characterization work, post irradiation examinations, safety testing, fission product release evaluation, as well as the measurement of chemical and thermomechanical material properties in representative conditions will feed a fuel material data base, applying strict QA enforcement. Further development of physical models enables assessment of in-pile fuel behavior under normal and off-normal conditions.

Fuel cycle back-end encompasses spent fuel treatment and disposal, as well as used graphite management. An optimized approach for dealing with the graphite needs to be defined. Although a once-through uranium cycle is envisioned initially, the potential for deep-burn of plutonium and minor actinides in a VHTR, as well as the use of thorium based fuels will be accounted for as evolutionary steps towards a closed cycle.

To answer these questions, an international collaborative program has been set up between the US, Japan, Korea, the European Union and France, under the GIF umbrella. The "VHTR/Fuel and Fuel Cycle" (VHTR/FFC) Project Arrange–ment (PA) became effective January 30, 2008, although the collaborative work had already started somewhat earlier (Figure 1). The present paper outlines the current status of the collaboration.

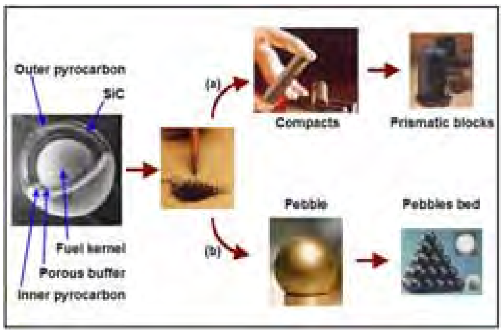

Figure 1: The VHTR particles fuel and the two types of fuel elements [a – Compact (courtesy of General Atomics), b - pebble (courtesy of JRC)].

II. THE VHTR/FFC PROGRAM AND ORGANIZATION

The VHTR/FFC research plan has been edited in 2007 and undergoes revisions as necessary. [2] For its elaboration, inputs from major industrial HTR and VHTR projects, such as PBMR, GTHTR300C, ANTARES, NHDD, GT-MHR, NGNP and HTR-PM, led by several plant vendors and national laboratories, have been considered.

The R&D plan is structured in work-packages and tasks as described in Table I.

Work package	Task
Irradiations and PIE	1.1 - Irradiation devices and procedures
	1.2 - Shared irradiation tests
	1.3 - PIE protocol and procedures
	1.4 - Irradiation and PIE results
Fuel Attributes and Material Properties	2.1 - Critical material properties
	2.2 - Fuel material property database
	2.3 - Characterization techniques
	2.4 - Fuel performance modeling
Safety testing	3.1 - Pulse irradiation testing
	3.2 - Heating test capabilities
	3.3 - Heating test
	3.4 - Source term experiments
Enhanced and Advanced Fuel	4.1 - Process development
Waste Management	5.1 - Head-end processes
	5.2 - Graphite management
	5.3 - Disposal behavior and waste package
Other Fuel Cycle Options	6.1 - Plutonium burning and transmutation
	6.2 - Thorium cycle

Table I: Structure of the VHTR/FFC research program.

In order to preserve the partners' intellectual property rights, there will be no open collaborative work on fabrication processes. For the time being, the process development to fabricate innovative TRISO fuels (such as ZrC coating process) remains out of the scope of the program as well.

From this research plan, the parties derive a bi-annual "action plan" that contains detailed descriptions of contributions and a list of deliverables with due date. The main milestones are:

- Irradiation and PIE (post-irradiation exami–nation)
  - 2015: irradiation PIE results
- Fuel attributes and material properties
  - 2009: Establishment of fuel material property database
  - 2009: Characterization techniques of fuel attributes and fuel performance modeling
- Safety testing
  - 2012: Pulse irradiation testing, esta–blishment of heating test capability, and source term experiments
  - 2015: Heating tests
- Waste management
  - 2010: Disposal behavior and waste package
- Other fuel cycle options
  - 2010: Plutonium burning and trans–mutation and thorium cycle assessment.

The Project Management Board (PMB), which meets twice a year, keeps track of the plan with the help of the OECD/NEA who, in particular, maintain a dedicated web site archiving all documents including the deliverables. The tracking includes budget and cost elements for each party both for current contribution and background information shared in the frame of the project.

III. STATUS OF ON-GOING ACTIVITIES

During 2008, the first "Action Plan" has been established covering the period 2007-2009, and identifies more than one hundred deliverables with the vast majority of which are associated with the two first work packages.

III.A. Irradiation and PIE

Numerous fuel irradiation tests had been conducted in Europe since the 1970s, in particular in support of the German HTR program. After a pause of about 10 years, such irradiations were resumed in the HFR Petten in 2004. Results from these most recent irradiations will be made available to this project as background proprietary information. It laid the foundations to more detailed (as opposed to systemic) experiments aiming to better seize specific material properties which are crucial for the understanding and correct modeling of fuel performance.

In this sense, the PYCASSO-I (*PYrocabone irradiation for Creep And Shrinkage/Swelling of Objects*) test is intended to generate basic thermo-mechanical properties of pyrocarbon under irradiation. Samples were provided by Japan, Korea and France. Samples from the US could not yet be included for scheduling reasons. The irradiation started on April 18, 2008, in the Euratom/JRC HFR reactor of Petten. It has proceeded without unexpected transients or changes observed. However, due to technical problems with the HFR reactor, the irradiation was suspended for several months in August 2008. It was resumed in February 2009 and is on track for completion in the second quarter of 2009. The PYCASSO-II experiment targets a higher fluence, up to 3×10^{25} n.m^{-2}, and is expected to begin irradiation in the second quarter of 2009 for 9 cycles of irradiation. Results from these irradiations will be part of work package 2 (Fuel attributes and material properties).

Because of technical problems with safety instrumentation and extended HFR downtime, the HFR-EU1 irradiation will continue into 2009. HFR-EU1 consists of 3 GLE4 pebbles and 2 pebbles produced by INET, and is intended to test high burn-up fuel performance in particular with respect to fission gas release.

The AGR-2 experiment is part of the general US/DOE VHTR fuel development program and is planned for the Advanced Test Reactor of Idaho National Laboratory. It will follow the AGR-1 irradiation which was the first one to test, in representative NGNP conditions, US fabricated fuels. Due to very good behavior of these fuels, the AGR-1 irradiation is being extended beyond plan so as to achieve a burn-up of 19.6% FIMA. Thus, the AGR-2 program has been delayed by a few months. It will carry both UCO (uranium oxicarbide) based TRISO fuel elements and classical UO_2 (uranium oxide) ones. Design work and fuel fabrication activities have been done with fuel being fabricated by the U.S., France and South Africa (as an invited partner). The irradiation experiment is anticipated to start in late 2009.

In parallel and following several workshops on this topic, intense work is being performed by all partners to provide procedures for post-irradiation examinations and to set up the equipment. The sharing of these procedures and methods ensures improved comparability, replicability and quality of the result.

III.B. Fuel Attributes and Material Properties

The objective is to compile material property needs, to appreciate their importance in regard to known fuel failure mechanisms and to obtain state-of-the-art material properties data. A prioritization of the properties to obtain an optimized testing/measurement plan with the objective to define the details of the proposed analytical irradiation (such as PYCASSO), and potential other irradiations has been defined. Workshops have been held for discussing details of the experiments and the samples that will be provided.

Dedicated experiments have been set-up to measure basic properties of the materials used in the TRISO fuel concept (Figure 2). Reports on specific issues (SiC under irradiation, thermal diffusivity measurement etc…) have been completed and have been issued in early 2009.

Figure 2: Equipment used at DOE/ORNL to measure strength of SiC and PyC.

Regarding fuel performance modeling (illustrated in Figure 3), a round robin test has been run in the frame of the IAEA CRP (-6) (Coordinated Research Program). Modeling mainly focuses on failure fraction determination and fission product release. All FFC members have participated in this work and the results are expected to be made available to the project after 2009 as the CRP-6 is extended to the end of 2009.

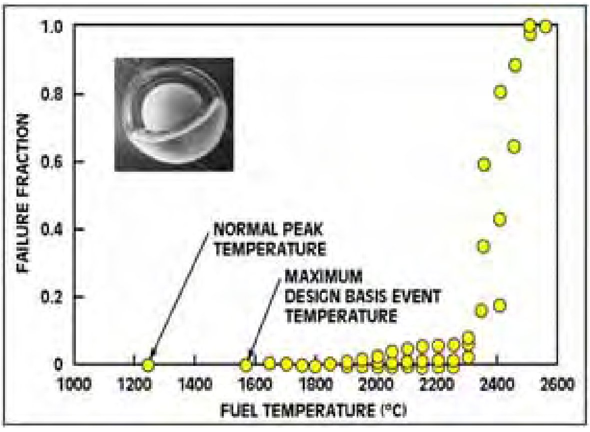

Figure 3: Influence of temperature on TRISO fuel failure fraction.

III.C. Safety testing

Safety testing comprises two types of tests on irradiated fuels: pulse irradiation and high-temperature (up to 2 000°C) heating experiments.

Post-irradiation heating tests can be performed in either of two ways:

- In a furnace/autoclave with a sweep gas transporting the effluent downstream to a thermal gradient tube in which condensable fission products are deposited, and to a final filter/gas trap to capture the non-condensable fission products. This approach has been used historically with LWR fuel. The advantage of the thermal gradient tube is that with precise measurements, the chemical form of the fission products can be inferred by the deposition profile.

- In a furnace with a sweep gas transporting the effluent to a cold finger that traps all fission products. The cold finger apparatus is then assayed using gamma spectroscopy to determine the fission product content. This approach has historically been used in the gas reactor community. No inference about fission product chemical form is possible with this approach.

Some parties (EURATOM, Japan, France) have already the equipment and, eventually, have run several tests (EURATOM, Japan). Reports are expected to be issued shortly. The current work of other parties focuses on establishing or up-grading heating test capabilities as well as the definition of test protocols.

III.D. Enhanced and Advanced Fuel

Although innovative processes has to be developed and qualified to produce advanced VHTR fuels (such as the replacement of the SiC layer with ZrC-zirconium carbide), there is currently no activity on this subject within this project.

III.E. Waste Management

This domain covers two issues:

- Spent VHTR fuel management
- Irradiated graphite management

For the spent fuel, the research activity currently deals with long-term repository/direct disposal for SiC particles. It is believed that the SiC coated fuel particle acts as a miniature containment vessel to retain fission products during long-term repository or direct disposal. Confirmatory tests, which prove the long-term integrity of the coating layers, are needed. This work is underway in the frame of the EURATOM project RAPHAEL which will finish in April 2010. First reports have already been issued.

Another interesting route to investigate is the reprocessing of fuel which would contribute to a significant reduction of waste volumes and potential radio toxicity in comparison to the direct disposal. The main issue is there to access the particle kernels for dissolution and the work currently focuses on that. First results, obtained by EURATOM are encouraging. Contributions from JAEA and US/DOE in this field are also expected.

Regarding graphite management, besides the establishment of a detailed inventory of graphite waste produced by a VHTR, the work deals with technologies to separate the highly active fraction from low-activity and to evaluate the feasibility to reuse the graphite. Main contribution will come from the Euratom CARBOWASTE project which will last until April 2012.

III.F. Other Fuel Cycle Options

VHTR can also be used with Pu fuel and for Minor Actinide (MA) incineration or trans–mutation, due to the high burn-up capabilities of coated particle fuel. These features can also be used in symbiosis with other reactor types to reduce MA content and decay heat which are decisive parameters for repository design. The deep-burn potential of VHTR avoids multi-recycling of spent fuel as it is needed for alternate routes. It is especially attractive if it can be shown that ultra-high burn-up coated particles are still capable to maintain their barrier function under disposal conditions.

Currently, the activities in the domain are conducted in the frame of the EURATOM PUMA project, which will be concluded in summer 2009 and by the US/DOE with the "Deep-burn project".

The PUMA project deals with Pu/MA HTR burner reactor physics & optimization, trans-uranics bearing fuel design and manufacturing, assessment of the impact on fuel cycle and economics, and the qualification of analysis tools. Formal agreement from PUMA contributors is pending to allow the results to be incorporated in the FFC project.

The US "Deep burn" project has started in July 2008 under the leadership of INL. It covers two main areas of research:

- Fuel Cycle analysis (core design, fuel performance in reactor, repository issues, assessment of fuel cycle scenarios …),
- Fuel Development (design of TRISO fuel, fabrication, recycle technologies …).

No activity has yet started within the FFC project regarding the assessment of the VHTR thorium fuel cycle.

IV. CONCLUSION

After one year of collaborative work, the Fuel and Fuel Cycle project of the VHTR is producing its first results. Despite initial difficulties to protect intellectual property of the partners, all parties have succeeded to join their effort in an almost comprehensive program covering all aspects of fuel development and qualification and waste management issues.

References

1. *Coll*., Fuel & Fuel Cycle Project Plan of the Very High Temperature Reactor system, OECD/NEA edition (2007).
2. *Coll*., Project Arrangement on Fuel and Fuel Cycle For the International Research and Development of the VHTR, OECD/NEA edition (2007).

Glossary

AGR	Advanced Gas Reactor
ANTARES	AREVA New Technology based on Advanced gas-cooled Reactors for Energy Supply
FFC	Fuel and Fuel Cycle
GTHTR	Gas Turbine High Temperature Reactor
GT-MHR	Gas Turbine Modular High temperature Reactor
HFR	High Flux Reactor
HTR-PM	High Temperature Reactor-Pebble bed Modules

LWR	Light Water Reactor
NGNP	Next Generation Nuclear Plant
NHDD	Nuclear Hydrogen Development and Demonstration
PBMR	Pebble-Bed Modular Reactor
PIE	Post Irradiation Examinations
PMB	Project Management Board
PUMA	Plutonium and Minor Actinide management
PYCASSO	PYrocarbon irradiation for Creep And Shrinkage/Swelling of Objects
RAPHAEL	ReActor for Process heat, Hydrogen And ELectricity generation
SiC	Silicone Carbide
TRISO	TRIstructural ISOtropic
VHTR	Very High Temperature Reactor
ZrC	Zirconium Carbide

STATUS OF ONGOING RESEARCH AND RESULTS: HYDROGEN PRODUCTION PROJECT FOR THE VERY HIGH TEMPERATURE REACTOR SYSTEM

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I. INTRODUCTION

High temperature processes for large-scale production of hydrogen are being investigated as potential uses of process heat from the Very High-Temperature Reactor (VHTR) system. Hydrogen currently has a large market worldwide for fertilizer production and in crude oil refining. Future applications for hydrogen are seen in increasing use of fuel cells, in production of alternative liquid fuels, and in production of chemicals. Additional benefits of hydrogen production from nuclear energy include higher efficiency and reduced green-house gas emissions than the currently predominant production methods.

The VHTR Hydrogen Production Project is beginning to compile the results obtained to date and provided to the project by the member countries. Working groups of technical experts are being organized to focus cooperative efforts on specific topics. Areas of cooperation include: developing and optimizing the thermo-chemical water splitting processes of the sulphur family, giving priority to the sulphur-iodine (S-I) cycle; advancing the high-temperature electrolysis process; evaluating alternative thermo-chemical hydrogen-generation processes (including processes amenable to operation with other Generation IV reactor systems); and defining and validating technologies for coupling reactors to process plants. Progress in these areas will be described in this paper.

II. DEVELOPMENT OF THE SULPHUR-IODINE (S-I) CYCLE

This portion of the project focuses on the evaluation of the Sulphur-Iodine (S-I) thermo-chemical cycle for H_2 production, which is one of the potential processes for large-scale deployment and coupling with the nuclear VHTR. The S-I process has been chosen as a reference amongst the multiplicity of alternate thermo-chemical cycles because it exhibits the best prospect regarding efficiency. The S-I process is illustrated in Figure 1. Acquisition of reliable thermodynamic data for the three basic reactions of the S-I thermo-chemical process is essential to assessing its potential for hydrogen production, as well as to determining operating parameters and estimating the cost of hydrogen production. In the S-I cycle, iodine and sulphur dioxide are added to water in an exothermic reaction that creates sulphuric acid and hydrogen iodide. The sulphuric acid can be decomposed at about 850°C, releasing oxygen and recycling sulphur dioxide. The hydrogen iodide (HI) can be decomposed at about 450°C, releasing hydrogen and recycling iodine.

(1)	$2H_2O + SO_2 + I_2 \rightarrow 2HI + H_2SO_4$	100°C	(exothermic)
(2)	$H2SO4 \rightarrow SO2 + H2O + ½O2$	850–900°C	(endothermic)
(3)	$2HI \rightarrow I2 + H2$	400–500°C	(endothermic)
	$H_2O \rightarrow H_2 + ½O_2$		

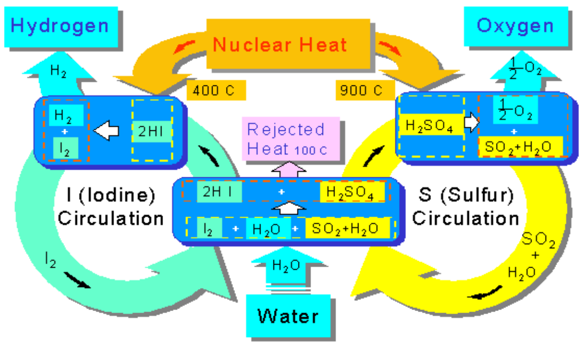

Figure 1: Sulphur-Iodine process

Several members are providing flow sheet analyses of the S-I cycle. These analyses are planned to be synthesized into a combined overview of the state of the art by the end of 2010. Benchmark exercises on a reference flow sheet are also planned to be performed. Several members are providing results of materials screening which has been performed via coupon tests and autoclave tests in environments simulating the different sections of the S-I process. Other materials screening and development activities involve membranes and adsorbents for separations, and catalysts for SO_3 and HI decomposition.

Interested members have progressed to performance of component and closed-circuit bench-scale experiments at full temperature, pressure, and flux rates to define and evaluate key parameters such as thermodynamic properties, rate constants. These activities are expected to be conducted over the next couple of years by various members to obtain additional experience with scaling up the process and constructing components with engineering materials. For future planning, interest has been expressed in international collaboration on pilot-scale plant construction and performance tests to confirm scaling parameters and materials performance.

III. DEVELOPMENT OF HIGH-TEMPERATURE ELECTROLYSIS (HTE) PROCESS

High temperature electrolysis (HTE) is one of the promising methods of producing hydrogen from nuclear energy. The technology and materials for a high temperature electrolytic cell is similar to that being developed for the solid oxide fuel cell program. The solid oxide electrolytic cell (SOEC) as being developed in current programs requires temperature in the range of 750 to 900°C for optimum efficiency. The energy content in the high temperature steam reduces the electrical energy requirement for the electrolysis, resulting in an overall efficiency improvement over conventional electrolysis. The HTE R&D program will focus on the production of hydrogen from the VHTR, with a core outlet temperature in the range of 900 to 950°C. It is anticipated that future work will also include examination of techniques for extending the temperature range of the HTE hydrogen production methods to other Generation IV reactor systems. Since HTE splits water in a device very similar to a solid oxide fuel cell (SOFC), the results of several national programs for electricity production from fuel cells will be monitored to assure the progress in SOFC technology provides key developmental data for the HTE program.

The electrochemical reactions taking place in the solid oxide cell are shown in Figure 2. An inlet stream containing steam at 800-830°C, plus about 10% hydrogen to maintain reducing conditions, is introduced to one edge of the cell. The water molecules are dissociated at the electrode-electrolyte interface and the oxygen is transported as $O^=$ ions through the electrolyte. A mixture containing about 90% hydrogen and the residual steam exits from the opposite edge of the cell. Oxygen molecules are formed at the electrolyte-anode interface and exits from the cell through flow fields adjacent to the anode. In reality, the oxygen flow fields are perpendicular to the place of the diagram, such that the oxygen and hydrogen are flowing are right angles to one another.

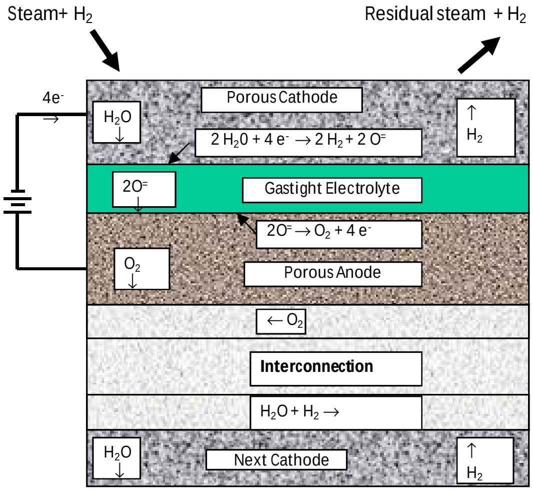

Figure 2: HTE processes in a High Temperature Solid Oxide Electroysis Cell.

Modeling activities for the HTE process have included optimizing system design for various plant configurations, examination of cogeneration options, and analyses of performance of cell configurations. Tests of button cells and small stacks of "standard" cells were conducted to investigate performance and longevity issues. In 2008 a HTE integrated laboratory-scale experiment was operated at 15 kWe with an initial hydrogen production rate of over 5000 liters per hour. However, over a two month period of operation the electrolyzer performance degraded significantly. Current efforts are focused on identifying the causes of cell degradation and performing tests of small stacks of cells. Three members are actively pursuing advancements in electrode materials, cell interconnect technologies, leak management solutions, and optimized operating conditions.

IV. ASSESSMENT OF ALTERNATIVE CYCLES AND ECONOMIC EVALUATION

Of the hundreds of methods for producing hydrogen that are available, only the S-I thermo-chemical cycle and high temperature electrolysis were agreed upon for initial collaborations under this project. Knowing that there was interest in various countries in other cycles, a work package was established to encompass technical evaluation of potential alternative cycles. Many cycles have been evaluated by several member countries with reference to S–I and HTE regarding methodology, feasibility and process efficiency and economics. Two of the cycles which have generated a great deal of international interest have been the Hybrid Copper-Chloride (Cu-Cl) cycle and the Hybrid Sulphur (HyS) cycle. Other cycles are being pursued as well to a lesser degree. Additionally, tasks involving economic evaluation of the various hydrogen production processes coupled to nuclear reactors are being performed.

Preliminary process development is proceeding for cycles of interest. In the case of HyS, there is a proposal to create a separate work package to focus additional R&D on that cycle.

V. COUPLING OF REACTORS AND ANY HYDROGEN PRODUCTION PROCESS

The final area of collaboration being pursued under this project regards analysis of the issues encountered when coupling hydrogen production processes to a nuclear reactor. Factors being considered are design-associated risk analysis, safety (including tritium abatement), and system integration. Performance calculations for interactions between the reactor and hydrogen plants are being evaluated in steady state to be followed by dynamic simulations. Work is beginning on coupling component technologies, such as process heat exchangers, high-temperature isolation valves, hot fluid ducting, and a thermal load absorber.

Figure 3 depicts a notional schematic of a VHTR, heat transfer loops, and coupling to the thermo-chemical and/or high-temperature electro-lysis plants.

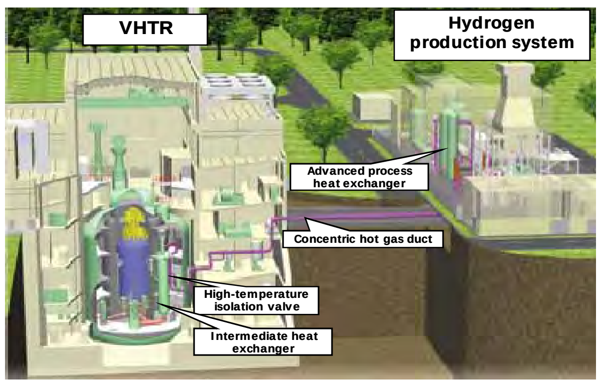

Figure 3: Artist's description of a hydrogen-production unit coupled to a very high-temperature reactor.

IV. CONCLUSION

Since the signing of the project plan in mid-2007, the VHTR Hydrogen Production Project members have been working to assemble and catalog their contributions of data which was generated prior to the formalization of the multilateral agreement. This process is anticipated to be complete in the spring of 2009. At the same time, the formal process for contributing deliverable reports for work completed within the scope of the project during 2008 and 2009 is being exercised in accordance with the GIF guidelines. Once these technical reports have been made available to the entire project members, plans call for workshops to draw summary conclusions and plan additional tasks to move the research forward or fill in gaps in the data as needed.

Acknowledgements

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STATUS OF ONGOING RESEARCH WITHIN THE GIF VHTR MATERIALS PROJECT

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I. INTRODUCTION

Expanded nuclear energy is a key element necessary to provide an adequate supply of clean, sustainably energy to meet the increasing demands of the world's expanding population and economy. This expansion will require a new generation of nuclear technology to augment the addition of evolutionary light-water-cooled reactors and life extension of the existing nuclear fleet. The Generation IV International Forum (GIF) has developed a technology roadmap for advanced nuclear energy systems that culminated in the selection of the six most promising Generation IV nuclear reactor systems that would best meet broad goals established for sustainability, economic competitiveness, safety and reliability, and proliferation and physical protection.[1]

Among the six advanced nuclear energy systems that were identified as contributing to the Generation IV goals was the Very High Temperature Reactor (VHTR), which employs a thermal neutron spectrum with coolants and temperatures that enable generation of high-quality process heat for hydrogen production or other commercial applications (such as those for the synfuel, petro-chemical, and steel industries), as well as electricity production with high efficiency.

Since the completion of the roadmap, GIF has coordinated worldwide developmental activities for Generation IV reactor systems. A Steering Committee has been formed for the VHTR to help plan and carry out the research and development (R&D), design, and safety studies being conducted by the participating GIF members to establish its viability and optimize its performance. Additionally, system-specific Project Arrangements (PAs) for key VHTR technologies, including structural materials, have been developed, which stipulate what specific international contributions to the advancement of those systems will be made and how information is to be shared. The VHTR Materials PA includes major contributions of both new and protected historical information from its participating partners that currently include Canada, EURATOM, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, and the United States.

Materials development and qualification, design codes and standards, as well as construction methodologies, for VHTRs require new investigations for the design and construction of the key components. The development of new material grades, as well as the extended qualification of existing materials, are key issues for meeting the higher temperature and longer lifetime requirements of VHTR normal and off-normal operating conditions, including:

- graphite for the reactor core and internals;
- high-temperature metallic materials for internals, piping, valves, high-temperature heat exchangers, steam generators, and turbo-machinery; and
- ceramics and composites (*e.g.*, C/C,

SiC/SiC, etc.) for control rod cladding and other specific reactor internals, as well as for advanced intermediate heat exchangers for very-high-temperature conditions.

II. GIF VHTR MATERIALS PROGRAM

The key design parameters that will affect the choice of materials and, therefore, the needed R&D include the reactor coolant inlet and outlet temperatures and pressure, as well as the choice of the secondary-side coolant and its associated temperatures and pressures. Expected service conditions include a near-term core coolant outlet temperature between 750 and 900°C, for which existing materials may be used, and a longer-term goal of 1 000°C that will require the development of new materials. The inlet core temperature for such systems could range from about 300°C to 600°C and the primary coolant system pressures from 5 to 9 MPa. Reactors currently being developed, such as the Next Generation Nuclear Plant (NGNP)[2], the Pebble Bed Modular Reactor (PBMR)[3], or the High Temperature Gas-cooled Reactor-pebble-bed Module (HTR-PM)[4], focus on the lower range of core outlet temperatures and will largely utilize existing structural materials, but will serve a vehicle for developing and evaluating the enhanced materials codes and design methods, as well as condition monitoring techniques, needed for their anticipated 60-year lifetimes and design envelops.

To efficiently coordinate the materials development and qualification activities within the VHTR Materials PA, a detailed Project Plan (PP) has been developed under the guidance of VHTR Materials Project Management Board (PMB) that is anticipated to be formally approved and implemented early in 2009. The PP includes three work packages that cover experimental and analytical activities on graphite, high-temperature metallic materials and design methods, and ceramics and composites being conducted from 2007 through 2012 by all partners, as well as identifying their contributions of protected historical information. Deliverables in each of these three areas include both individual technical contributions from the GIF partners (*e.g.*, individual sets of data on mechanical or thermo-physical properties for a particular grade of graphite) and multinational products (*e.g.*, a joint report summarizing experimental data and analysis of the micro-structural stability of Ni-base super alloys in a VHTR helium environment). Contributions with a total value of well in excess of $ 200 M have been identified by the signatories to the VHTR Materials PA.

Materials working groups, comprising technical experts from each GIF signatory, are responsible for coordinating the input to each of the three work packages and advising the VHTR Materials PMB on the technical sufficiency and monetary value of the contribution from each signatory to ensure appropriate progress is made and shared by all partners. Annual review of all work plans and contributions will be made.

III. VHTR GRAPHITE STUDIES

The graphite components of the reactor include the permanent inside and outside reflectors, the core blocks, and the core supports. New graphite grades that are anticipated to show good performance under VHTR in-service conditions are being procured. New fine-grained isotropic graphite types with high strength and low irradiation damage are required to achieve high outlet-gas temperature, long life and continuity of supply. Extensive irradiation and properties test data are needed to qualify the new materials. The reference materials for the side reflectors and core support blocks may be UCAR PCEA or SGL NBG-17 or NBG-18 graphite grades, though several other graphite grades are being considered. At the current time, NBG-18 has been selected for the South African PBMR and IG-110 has been selected for the Chinese HTR-PM, as well as the Japanese GTHTR300C, reactors. Either PCEA or NBG-17 is suitable for use in prismatic reactors, but no vendors or other VHTRs have selected either of these grades at this time.

Participating signatories of the VHTR Materials PA are coordinating the acquisition, management and traceability of candidate nuclear graphites to optimize their overall graphite qualification activities and data generation needed for mechanical, thermo-physical and fracture properties. Such data are being developed as a function of temperature from 25-1 600°C, as well as for graphite oxidation kinetics and the effects of oxidation on relevant mechanical and physical properties in both He-coolant and air. The variations of properties with specimen volume, orientation, position within billet, between billets, and between lots are being addressed.

Effects of neutron irradiation on dimensional changes and properties are being assessed. Data will also be generated for the irradiation-induced creep rate and creep coefficients over relevant dose and temperature ranges. The mechanism of displacement damage in graphite via particle irradiation will be examined in comparative particle irradiation studies to elucidate the differences in behaviour of various graphite grades. Mathematical and mechanistic models are needed to allow interpolation and extrapolation of irradiation effects data. Hence, models for irradiation-induced dimensional changes, thermal conductivity, strength, fracture behaviour, and irradiation-induced creep are being developed, as are stress analysis codes and finite element models for modelling the stress states in components and predicting failure. A particularly valuable example of collaboration among GIF partners is provided in Figure 1, where coordinated individual contributions of irradiation experiments to meet design requirements are collectively displayed.

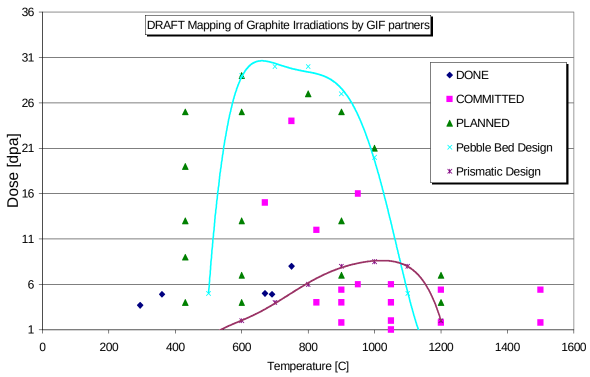

Figure 1: Comparison of graphite irradiations by GIF partners with design needs.

It can be seen that the operating temperatures for graphite in prismatic designs extends to a slightly higher range than for pebble bed designs, but the biggest difference in operating conditions between the two designs is the much higher irradiation dose to which graphite immediately adjacent to the pebbles is subjected. To address the collective set of data needs for both designs, participants in the VHTR Materials PA have jointly agreed to develop data that will cover full range needed. Early results will address the lower doses anticipated for the prismatic designs, since very long irradiation exposures are required to reach the highest doses typical of pebble bed operation.

Consensus design codes [*e.g.*, American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME) and Japan Society of Mechanical Engineers Codes for Power Facilities (JSME)] are needed for graphite core structures and consensus test methods [*e.g.*, ASTM International and International Organization for Standardization (ISO)] are needed for nuclear graphite property determinations. The ASME and JSME have begun to establish such design codes including rules for materials selection and qualification, design, fabrication, testing, installation, examination, inspection, and certification and to prepare reports guiding manufacture and installation of non-metallic internal components for fission reactors. Development of ASTM standards for nuclear graphite materials specifications and a wide range of mechanical, thermo-physical, and fracture testing standards is underway. Existing French and German [Deutsches Institut für Normung (DIN)] standards are being also being updated and adopted as ISO standards. Participation in these codes and standards developments is an active component of the GIF VHTR materials program.

IV. VHTR METALS AND DESIGN METHODS STUDIES

Metallic materials will be needed for several reactor sub-systems including: the reactor pressure vessel, high-temperature metallic core internals, hot ducts and other pressure boundary components for the primary coolant system; the

Intermediate Heat exchanger (IHX) coupling the reactor to secondary systems; and the turbo-machinery, heat recuperators, and steam generators used for electric energy production.

This R&D program has been defined according to service temperature conditions.

- Low-temperature materials T<650°C for the reactor pressure vessel and other structural parts, including both qualification of materials typically used in LWR systems under VHTR system conditions, as well as higher temperature alloys suitable for service under conditions where time-dependent processes such as creep and creep-fatigue are significant.

- High-temperature materials, notably for metallic reactor internals, intermediate heat exchangers, and steam generators.

Characterization of materials and welds will be performed for relevant service conditions for each class of materials.

- High-temperature mechanical properties (*e.g.*, tensile, creep, creep fatigue, stress-rupture, high and low-cycle fatigue, fracture toughness) in both air and impure helium environments, as well as following irradiation exposure.
- Environmental degradation processes from exposure to high-temperature helium with contaminants such as CO, CO_2, H_2, H_2O, and CH_4.
- High-temperature metallurgical stability (*i.e.*, thermal aging effects).

Much of the R&D to be performed under the current GIF VHTR Materials PA will be for service conditions between about 350°C<T<900°C and will focus on traditional LWR low-alloy pressure vessel steels, such as A533B and A508, under the longer times and slightly higher temperature required for VHTR pressure boundary applications, as well as super alloys, such as Hastelloy X, 800H, IN617, and Haynes 230 for IHX, steam generator, and high-temperature internals applications. Longer-term R&D will include both development of materials for very high-temperature service beyond 900°C (*e.g.*, oxide dispersion strengthened alloys, refractory-based and advanced super alloys), as well as the qualification of existing materials for nuclear service at intermediate temperatures, such as modified 9Cr-1MoV for higher temperature pressure vessels.

At the current time, LWR pressure vessel steels (*e.g.*, A533B and A508) have been selected as the pressure vessel material of choice for all the VHTRs currently under development. Concerns about availability of large forgings, and the still unproven commercial capability to fabricate the very large ingots they require without macrosegration from advanced pressure vessels steels such as modified 9Cr-1MoV, have led to engineering approaches (*i.e.*, vessel cooling or insulation) to ensure the operating vessel temperatures are low enough to use the LWR steels. The greater challenge for the LWR steel use is their ability to survive potential short-term high-temperature excursions related to loss of coolant flow. Some studies have indicated that the current time-temperature limits for A533B and A508 in ASME Code Case N-499 may be exceeded during such transients. Moreover, the operating temperatures limit of 371°C assumed for time-independent ferritic steel operation may require additional assessments and/or justification of potential creep and creep-fatigue effects to reach the 600 000 hour operating lifetime desired for the vessels.[5]

The research on modified 9Cr-1MoV and its associated weldments illustrate significant synergism in the R&D among signatory members. Many different aspects of this material's behavior and design methods have been addressed by different signatories, resulting in a collectively developed compendium of data and design methods that are critical to its deployment in potential VHTR designs. These efforts include:

- tensile and cyclic mechanical properties;
- creep, creep-fatigue and creep crack-growth data;
- fracture and charpy impact toughness;

- environmental effects due to impure helium, oxidation, and irradiation;
- microstructural evolution and thermal aging for long-term service;
- allowable design and operational stresses, and
- design rules for creep-fatigue interaction and negligible creep conditions.

Candidate materials for IHX and steam generator applications must have a combination of high-temperature strength and corrosion resistance in the impure helium typical of gas-cooled reactor environments. Wrought high-Ni creep-resistant alloys containing 20 to 22 wt% Cr are creep resistant and offer protection against oxidation up to about 900°C by formation of chromia scale, however none are fully qualified in ASME code for HTGR nuclear applications and will require additional qualification data. Only Alloy 800H, is currently ASME Code qualified for high-temperature nuclear service and then only to 762°C.

	Ni	Cr	Mn	Co	C	Fe	Ti	Al	W	Si	Mo
Inconel 617	base	22.0	0.40	12.0	0.10	2.0	0.40	1.2	-	0.40	9.0
Haynes 230	base	22.0	0.65	5.0	0.10	3.0	-	0.30	14.0	0.50	2.0
Alloy 800H	32.0	21.0	1.00	-	0.06	bal	0.40	0.40	-	0.60	-
Hast X	base	22.0	1.00	1.50	0.10	18.5	0.15	0.50	0.60	1.00	9.0

Table 1: Composition of principal high temperature alloys for VHTR IHX and steam generator applications.

Inconel 617 and Haynes 230 are the leading candidates for application above 800°C, as they have greater strength at these temperatures. Haynes 230 appears to have slightly higher corrosion resistance in VHTR helium environments, but the much greater database for Inconel 617 and the existence of a well developed draft ASME Code case have led most designers to favor the use of 617 for higher temperature applications. Below 800°C, the lower cost and Code status of Alloy 800, are advantageous. A special variation of Hastelloy X developed by the Japanese (Hastelloy XR) with tighter controls on some alloying elements appears to offer greater environmental resistance to VHTR He, and has been used in the Japanese High Temperature Test Reactor (HTTR) IHX at operating temperature of 950°C for short periods, and is the current choice for the IHX of their advanced GTHTR300C reactor.

Another area that illustrates the benefits of the combined work from the signatory members is in the R&D efforts on Alloy 617. It is a candidate material for the very high temperature metallic components such as the intermediate heat exchanger and the hot ducting in potential VHTR designs. Progress has been made in the following areas:

- tensile and cyclic mechanical properties;
- fatigue, creep rupture, and creep-fatigue data;
- environmental effects due to impure helium, oxidation, and irradiation;
- microstructural evolution and thermal aging for long-term service;
- implications of deformation mechanisms for long-term service conditions; and
- viscoplastic constitutive models to support design analysis methods.

V. VHTR CERAMIC AND COMPOSITE STUDIES

Ceramics and structural composites are regarded as backup or advanced solutions to metallic materials challenges for several VHTR components because of their superior high-temperature strength or radiation resistance. Key areas for collaborative studies on these materials focus on their use for heat exchangers, control rods, insulation materials, and internals structures such as restraints and fasteners.

V.A. Structural Composites

Carbon fiber reinforced carbon (C/C) composites with useable service temperatures up to 1 800°C and other ceramic composite materials (for example, SiC/C and SiC/SiC) have been proposed for the several internal subcomponents in the near term and for the control rod assembly in the longer term. The C/C and SiC/SiC composite and ceramic materials are relatively new reactor materials for which irradiation and other material properties data are needed. Standardization and codification of materials from within these industries are also major issues that will need to be resolved for use of these materials in reactor safety-related systems.

Mechanical and thermal property, fracture behaviour, and other tests, including oxidation effects and post-irradiation evaluations as a function of fabrication methods are required to establish design guidelines and a design database. The modelling of the material behaviour and stress analyses in these codes will need to consider the anisotropic nature of these materials. Obtaining non-destructive testing data and fracture toughness data is necessary to establish acceptance guidelines.

V.B. Ceramics

High-temperature fibrous insulation may be used throughout the reactor and power conversion systems, notably in the hot duct, upper plenum shroud, shutdown cooling system, helium inlet plenum, and turbo-compressors. Ceramic insulation blocks may be needed under the graphite core support structure. Insulating materials that retain resiliency minimize off-gassing, and do not shed particulate under high gas flows and irradiation damage are needed. All these materials need to be fabricated, tested, and qualified for use under VHTR conditions. Qualification of non-metallic materials will require, in some cases, the development of recognized industry standards and codes for materials and testing.

Current activities include evaluation of composites for in-pile and out-of-pile components including advanced control rods, as well as for stabilizing straps and ties for the core. Evaluation of mechanical and thermal properties and the dimensional stability for both C/C and SiC/SiC composites for un–irradiated materials and at irradiation doses up to about 10 dpa for C/C and over 20 dpa for SiC/SiC composites is ongoing by multiple GIF members.

VI. MATERIALS MODELING

After the introduction of quantitative descriptions of creep and creep-damage mechanisms in metals in middle of the last century (*e.g.*, collective work by Norton, Kachanov, Monkman-Grant, etc.), it took about 25 years to develop a working engineering understanding of creep-fatigue interactions (*e.g.*, collective work by Manson, Coffin, Mowbray, etc.). The introduction of damage mechanics in terms of subcritical crack growth and the introduction of constitutive laws for creep-fatigue interactions (*e.g.*, Chaboche) was a further improvement in lifetime assessments of structures. With the current availability of huge computer clusters operating in parallel mode, numerical solutions of equations for atomistic behavior became very attractive. Although it is well accepted that damage starts at atomistic levels, it is not easy to bridge the gap between atomic and structure levels and requires an understanding of the related physical phenomena on a range of scales from the microscopic level all the way up to macroscopic effects.

Determination of the life-time of components exposed to severe environments such as in VHTRs is very demanding, particularly when damage interactions (like creep-irradiation or strength-microstructure, toughness-irradiation induced phases) must be considered. The simulation of materials behaviour under such extreme conditions needs to encompass broad time and length scales from atomistic descriptions of primary damage formation to a description of bulk property behavior at the continuum limit. This requires a multi-scale, multi-code modelling approach that begins at the atomistic level with *ab initio* and molecular dynamics techniques, moves through the meso-scale using reaction rate theory models, lattice kinetic Monte-Carlo and Dislocation Dynamics, and ends with the macro-scale using Finite Element methods and continuum models.[6] Experimental validation of the modelling results is mandatory. This approach is schematically shown in Figure 2.

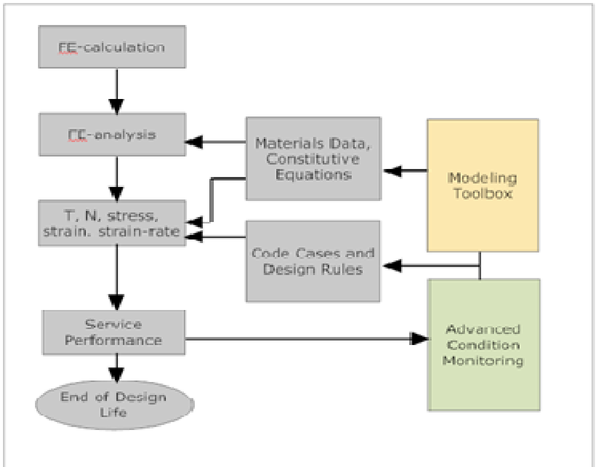

Figure 2: Illustration of approach for interactive support of materials modelling in reactor component design.

The necessary time required for the development of advanced Generation IV systems can provide the necessary lead time to allow such a multi-scale modelling approach to provide useful input to both qualification of existing materials and the development of newer, higher performance ones.

Towards that end, modelling activities will be conducted to support and interact with the experimental and technological part of the studies within the VHTR Materials PA. It is anticipated that this will strongly support the development of a mutual understanding of the design needs and solid-state physics necessary for improved assessments of long-term materials performance.

VII. METHODOLOGY & CODIFICATION OF HIGH-TEMPERATURE MECHANICAL DESIGN RULES

Cumulatively, the results of the materials R&D activities being performed under the VHTR Materials PA will provide input for the improvement of codes and standards needed for VHTR plants. Moreover, modelling and description of materials behaviour and damage development will provide an enhanced scientific basis for the codification work and damage assessments. Improvements of existing high-temperature design methodology, including structural design methods, materials testing and database developments, and nuclear design code and standards (such as the ASME or JSME Codes) must consider the:

- extension of design code approvals for metallic materials at higher operating temperatures and longer service lifetimes;
- development and approval of design code for graphite, composite, and ceramic materials in nuclear service.

Similarly, materials test standard development and approval to obtain qualified advanced materials properties by organizations such as ASTM or ISO must include the development of approved testing standards for thermal, physical, mechanical, and fracture properties of graphite and advanced composites.

VIII. MATERIALS DATABASE

The development of VHTRs requires extensive materials data on metals, graphite, ceramics, and composites. To efficiently manage the materials data and facilitate coordinating international activities, it was recognized that a materials property database that provides an authoritative single source and is internally consistent, validated, and highly qualified is crucial to the success of the program. Hence, a dedicated database has been developed and will be used to retain and assemble data provided by GIF members, data available in the literature, and other resources. It is web-based with highly secure access and will be used to coordinate the extent of testing with minimum redundancy; track and exchange results of various types of testing, test conditions, product forms, and metallographic information; assess and rank quality of test data; and preserve data for current and future use.

IX. CONCLUSION

To address the extensive needs for development and qualification of structural materials to support VHTR systems, a formalized international program for generation, exchange, and coordination of materials data has been established under the GIF framework. Within that framework, collaborative research on graphite, high-temperature metals, and ceramics and composites is being conducted. Results from this research are being exchanged among the participants and form the basis for augmenting the materials and design codes and standards needed for VHTR system deployment.

Nomenclature

ASME	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
ASTM	ASTM International
C/C	Carbon fiber reinforced Carbon Composite
GIF	Generation IV International Forum
IHX	Intermediate Heat exchanger
LWR	Light Water-cooled Reactor
VHTR	Very High Temperature Reactor
PA	system-specific Project Arrangement
PMB	Project Management Board
PP	Project Plan
R&D	Research and Development
SiC/C	SiC fiber reinforced carbon composite
SiC/SiC	SiC fiber reinforced SiC composite

References

1. U.S. Department of Energy, A Technology Roadmap for Generation IV Nuclear Energy Systems, Generation IV International Forum, GIF-002-00, (December 2002).

2 Corwin, W.R., *et al*., Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials, ORNL/TM-2008/129 (August 31, 2008).

3 Matzie, R. and G. Greyvenstein, Pebble Bed Modular Reactor Project Status, Presented at the 2009 International Congress on Advances in Nuclear Power Plants <www.icapp09.org/documents/material/PN4_2.pdf> (May 2009)

4. Zhanga, Z., *et al*., Current Status and Technical Description of Chinese 2 × 250 MWth HTR-PM Demonstration Plant, Nuclear Engineering and Design, Volume 239, Issue 7, pp. 1212-1219 (July 2009).

5. Jetter, R., *et al*., The Impact of Material and Design Criteria on the Assessment of Negligible Creep, PVP2009-77953 (July 2009).

6. Samaras, M., *et al*., Materials Modelling – A Key for The Design of Advanced High Temperature Reactor Components, Proceedings of ICAPP 2007, Paper 7207 (May 2007).

SESSION I SUMMARY / DISCUSSION

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SEssion I of the GIF symposium 2009 was composed of several presentations on R&D activities in two areas, the Crosscutting R&D Working Groups and the VHTR system. The presentations were focused on the major achievements made since R&D activities were launched and also, the expectations of achievement for the next five years.

Crosscutting R&D Working Groups

Three Crosscutting R&D Working Groups have been taking an active role in developing relevant methodologies for supporting the development of Generation IV systems. The methodologies developed by each WG were independently tested, evaluated, and then provided to each system to be used for system evaluation and development.

The EMWG (Economic Methodology Working Group) has developed a cost estimating guideline and verified a software package (G4-Econs) against benchmark models. Two approaches for the cost estimation were adopted for application to different levels of system design. The methodology was then applied to the JSFR for the purpose of analyzing its capability by comparing it with various different methodologies and to prove its reliability. The EMWG released the SW package for use by GIF SSC, IAEA, universities and the public for evaluation of the cost of their systems and to receive feedback from users.

The PRPP WG (Proliferation-Resistance and Physical Protection Working Group) has developed the PRPP methodology by identifying and implementing several important elements to be considered concerning the PRPP evaluation. Workshops and interactions with other spheres of activity contributed to further define the methodological approaches and the needs of users. The Safeguard by Design (SBD) concept is recommended as a mechanism for proliferation risk reduction assessment. Efforts are on-going to seek harmonization and the potential for synergy between GIF-PRPP methodology and INPRO initiatives.

An Integrated Safety Assessment Methodology (ISAM) for the safety evaluation of Generation IV systems was developed through the tremendous effort of the RSWG (Risk and Safety Working Group). Five principle postulates for an integrated philosophy are established and implemented into the methodology. The methodology also defined three design attributes for achieving safety goals. The PSA-based ISAM can be used at any stage of concept development and during the design phases of the system. The ISAM is not intended to measure the level of safety, but to contribute to achieving safety objectives during design development.

Discussions

The economic aspect is a fundamental element of importance together with the safety aspect in evaluating the merits and/or demerits of the system and associated technologies. Therefore, the highly reliable estimating capability of the economic evaluation tool is held in high confidence by users. It is thus advisable that the EMWG considers various mechanisms for promoting further broad interactions with other national and international projects and also performs case studies of various advanced nuclear system developments as ways to enhance the reliability and trustworthiness of the methodologies.

The PRPP is considered as a very difficult and challenging area with respect to getting consensus and agreement from the associated stakeholders. Various different strategic understandings among stakeholders may exist in dealing with the methodology and in interpreting the results of the analysis of the PRPP applications. The reliability and applicability of the methodology may be strongly dependent on the perspectives of the stakeholders of the PRPP. Close communication and interaction might be needed for increasing the common understanding and harmonized agreement within the society of the stakeholders.

Very High Temperature Reactor (VHTR) System and Technology

Several on-going national mid-term projects of the GIF member countries for the period of 2015-2025 were introduced and emphasis was placed on strengthening future R&D efforts to resolve various technical issues and to advance technologies. There are three R&D projects in the VHTR system currently on-going through collaboration within participating member countries: the VHTR Fuel and Fuel Cycle Project, the Hydrogen Production Project, and the VHTR Material Project.

The technology development for the TRISO coated fuel and studies concerning the back-end closed fuel cycle management are the main R&D areas of interest in the VHTR Fuel and Fuel Cycle project. Major on-going activities in the project cover a wide spectrum of R&D areas through collaboration with participating member countries. The activities include: pyrocarbon irradiation tests (PYCASSO program) and AGR-2 experiments, measurements of basic properties of TRISO fuel materials and experiments for fuel performance modeling, R&D and tests for long-term direct disposal of coated particles and feasibility evaluations for the graphite reuse, and the deep-burn program (PUMA) for burning the Pu fuel and Minor Actinide (MA) in VHTR.

The Hydrogen Production project is currently at the stage of compiling data and results concerning hydrogen production and also the technology of the system coupling between the reactor and hydrogen production system. For the Sulphur-Iodine (SI) process, major progress has been achieved in the areas of material screenings, tests of component performance, and bench-scale experiments. Efforts are being pursued to scale-up the process and to further collaborate on a pilot-scale plant construction. The R&D efforts for developing technologies for the High Temperature Electrolysis (HTE) process have been focused on R&D for hydrogen production while increasing the core outlet temperature to around 950 degrees, and even higher. As potential alternative processes, the Cu-Cl cycle and hybrid sulphur (HyS) cycle were investigated. The economic evaluation for these various processes is being performed. The technology of coupling the hydrogen production system with a nuclear reactor is technology of importance with reference to the safety, reliability, and performance of the coupled system. The coupling technologies are also under evaluation.

The R&D efforts in the VHTR Materials project have been focused on the work packages of three different materials – graphite, high-temperature metals, ceramics and composite - for use in different components of the reactor system. As an on-going R&D activity, the graphite study work package selected several candidate materials satisfying the selection criteria, and those selected materials are under various irradiation tests for evaluation. The design codes and standards are also being developed for material qualification. For the metal study work package, R&D programs for the selected materials have been defined, and performance studies for those selected materials are underway. The studies for the ceramic and composite work package have been concentrated on major issues such as standardization and codification of materials along with mechanical and thermal evaluation efforts. Further, a dedicated web-based database has been developed for harmonized use in VHTR development.

The priorities concerning the VHTR system and technology development for the next five years will be the following activities.

- The VHTR viability phase will be completed by 2010 by optimizing the design features and operating parameters of the VHTR systems.

- Further assessment for the range of candidate applications of VHTR with variable core outlet temperatures will be carried out.

- High temperature process heat for various industrial applications is an important domain of study.

- For the hydrogen production work package, efforts will be pursued to resolve the feasibility issues (process, technologies), and the priorities concerning R&D needs and pre-industrial projects will be updated.

- For material work packages, efforts will be pursued to resolve the feasibility issues (qualification, manufacturing) for core and cooling systems, and there will be an update of R&D priorities.

- For the fuel work package, major efforts will be pursued to establish performance margins and FP source terms of TRISO fuel particles.

Discussions

It is agreed that much valuable achievements arising from R&D efforts for the VHTR system and technology have been made in all the currently on-going projects. Most of the participating members in the VHTR system are conducting near-term deployment national projects and are providing their valuable output to the joint collaborative project. The experiences and lessons learned over past years in various types of gas-cooled reactor technologies have also been very effectively utilized in establishing guidance for the direction of the current R&D efforts to develop and advance the technologies necessary for the VHTR system.

The goal and objective of joint collaboration for the Generation IV VHTR system is to develop a baseline model system with very high coolant outlet temperature and other associated technologies. From the viewpoint of the GIF philosophy of collaboration among participating member countries, it is thus important to consider how to effectively harmonize and utilize the contributions provided from those near-term national projects for developing a GIF baseline model. Also careful consideration should be made on how to share the commercial technologies, information, and experiences of participating members without infringing on intellectual property rights.

The computational methods development project, the launch of which is still under discussion between provisional members, is considered, in general, to be a fundamental and essential project for a reactor and its components development. The methodologies developed and proved through this R&D project must be utilized to assess, analyze, develop, and design the system. Although various elements and considerations need to be discussed and there should be a consensus on how to initiate this project, it is strongly recommended that a common understanding is reached as early as possible among the provisional participating members with respect to the important role of the project in system development.

One of the elements receiving increased attention and concern from the nuclear society as well as from non-nuclear societies when considering the use of nuclear energy is the reasonable and employable mechanisms for managing spent fuel. The VHTR system is recognized as a reactor system producing a tremendous amount of spent fuel due to its operating characteristics compared to the other Generation IV systems. Thus, it is highly recommended that much more emphasis is placed on this area, and that focused R&D considerations for spent fuel management must be taken into account when defining further R&D activities.

As discussed above, the development of the Generation IV baseline model technology may take advantage of information and outputs contributed from the near-term deployment of national projects. However, in contrast to the initial target for the coolant outlet temperature of the Generation IV VHTR system, most of those national projects design their reactor system with the outlet temperature at around 750 degrees which is a much lower temperature than the Generation IV initial target around 950 degrees and/or even higher. In other words, it seems that the target coolant outlet temperature has been lowered in comparison with the original Generation IV philosophy and objectives without generally agreed consensus within collaborative environments. In a certain sense, this may be an acceptable strategic approach to create more confidence in the VHTR system development by assuring the reliability of the technology through the relatively low-temperature HTR experiences. However, it is still strongly desired that the initial target temperature of VHTR should be aimed at as a technical challenge for the development of the Generation IV VHTR system.

The material availability and reliability in circumstances of very high temperature may become an Achilles' heel concerning the successful development and commercial realization of VHTR technology. Although most of the material R&D activities have been focused on those aspects so far, further strengthened and comprehensive collaborative efforts are needed to concentrate on material development and qualification that is practically applicable in very high temperature environments.

SESSION II

GAS-COOLED FAST REACTOR (GFR)
SUPER-CRITICAL WATER-COOLED REACTOR (SCWR)
MOLTEN SALT REACTOR (MSR)

Co-Chairs: Roland Schenkel and Anatoly Zrodnikov

GAS-COOLED FAST REACTOR (GFR): OVERVIEW AND PERSPECTIVES

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I. INTRODUCTION

The Gas-cooled Fast Reactor (GFR) system features a high temperature helium cooled fast spectrum reactor. It is associated to a close fuel cycle. The GIF Technology Roadmap [1] identified the GFR as a technology that associates therefore the advantages of fast spectrum systems for long term resources sustainability, in terms of use of uranium and waste minimization (through fuel multiple reprocessing and fission of long-lived actinides) with those of the high temperature (high thermal cycle efficiency and industrial use of the generated heat for hydrogen or industrial process).

The GFR is a fast neutron spectrum system that must be seen as a complement to the SFR deployment, which benefits from a more mature technology, with higher potential performance for a longer term industrial deployment. It uses the same fuel recycling processes. The GFR can also be seen as a sustainable version of thermal spectrum helium-cooled reactors (HTRs), which also benefit from a more mature technology, with fuel recycling and optimal use of mining resources. It uses basically the same technology.

This paper illustrates the technical progress achieved in the countries participating to the GIF effort on the GFR system.

II. GFR IN GENERATION IV

The GFRs development approach is to rely on technologies already used for the HTRs but with significant advances, in order to reach the objectives stated above. Thus, it calls for specific R&D beyond the foreseen work for thermal HTRs.

The main GFR design specifications as derived from the general objectives of Generation IV systems are:

- Use of gas as a coolant as a means of reaching high temperatures;
- Economic competitiveness by means of simplicity, compactness and efficiency;
- A robust safety demonstration, based on probabilistic safety assessment and defence in depth principles, and including severe accident management.

Additional design specifications of the GFR include:

- Fast neutron spectrum core with a zero (self-breeding) or positive breeding gain, with no or very limited use of fertile blankets in order to:
    - Generate as much fissile material as it consumes, with an optimal use of uranium;

- Have a fuel cycle fed with only depleted or natural uranium;
- Achieve homogeneous recycling of all actinides, in order to have no separation of plutonium from other actinides (proliferation resistance).

- Core plutonium inventory not exceeding 10 tons/GWe, in order to have a realistic reactor fleet deployment (in a few decades) and high fuel burn-up.

In the HTR the use of graphite increases the thermal inertia of the core, thereby limiting the maximum temperature during transients. On the other hand, GFR cores have relatively low thermal inertia; design features aimed at overcoming this apparent unfavourable feature include:

- A fuel element based on refractory materials and high thermal conductivity, with the ability to ensure radioactive material confinement up to very high temperatures.
- A primary circuit design based on upward core cooling and a moderate pressure drop for all the primary components and circuit involved in accident scenarios. One essential parameter for safety system performance is gas pressure. The primary helium is pressurized to 7 MPa under nominal conditions. A gas tight envelope enclosing the primary circuit has been added in order to limit the loss of pressure in case of primary loss of coolant. Maintaining high helium density allows the Decay Heat Removal system to rely on moderate pumping power and even on passive natural convection in some situations.

The fuel element is able to withstand high operating temperatures and transients associated with the poor heat capacity of the gas coolant. The main temperature limits are the following:

- An operating temperature, around 1 000°C, that provides a sufficiently ample margin to failure;
- A boundary temperature of 1 600°C below which fission products release is prevented;
- An upper temperature of 2 000°C below which the core geometry can safely be cooled down.

Concerning the objectives of ultimate waste minimization, proliferation resistance and natural resources optimization (zero or positive breeding gain), the major corresponding reactor design options are:

- No fertile blanket and multi-pass recycling of all actinides without separation;
- Loading of 1.1% of Minor Actinides (corresponding to self-recycling);
- A high density fuel with maximisation of actinide content;
- High core power density of about 100 MW/m3;
- A high core power unit of 2 400 MWth (for economic reasons);
- Mean overall core Burn-Up: 5% FIMA.

These high level objectives imply various additional secondary specifications such as minimization of the reactivity swing per cycle, or minimization of the core pressure drops for example.

III. GFR DESIGN OPTIONS

Reference option of the GFR is a 1 200 MWe reactor for electricity production.

A significant effort has been carried out since 2001 to propose a first consistent design of the reactor and its fuel. The GFR design is still evolving, however major design directions have been decided on, concerning the fuel, core materials, reactor architecture, and safeguard systems. The current reactor design reaches the initial set of performance:

- Self-generation of Plutonium in the core to ensure Uranium resources saving;

- Uranium fertile blanket free to reduce the proliferation risk;
- A limited mass of plutonium in the core to allow an industrial deployment of the fleet taking into account the initial restricted inventory;
- An ability to transmute long lived nuclear waste resulting from a recycling of the spent fuel, without lowering the other quoted performance;
- A high power conversion ratio (favourable for economics).

In parallel, the safety architecture was thought to cover the potential defects fitted to this system, thanks to the following elements:

- A fuel element that uses refractory materials and withstands very high temperatures;
- A gas voiding reactivity effect in the core naturally not significant;
- A current design allowing the decay heat to be removed in any accidental situations (pressurized or not, even in case of large primary break, including an additional single failure or multiple failures), thanks to different systems of moderate power supply and to a gas tight envelope.
- In addition, natural convection capabilities can be retained in most of the situations (including small primary break), leading to a real advantage in terms of Decay Heat Removal strategy robustness and progress-siveness. Thus, these situations can be managed in a passive way, including the total loss of electrical power.

Nevertheless, several technical fields are only partially covered today or they need to be optimized.

Fuel element:

At least two fuel concepts have the potential to fulfil the above requirements, that is: a ceramic plate-type fuel element and a ceramic pin-type fuel element. The reference material for the structure is reinforced ceramic, a silicon carbide composite matrix ceramic. The fuel compound is made of pellets of mixed uranium-plutonium-minor actinide carbide. A leak-tight barrier made of a refractory metal or of a Si-based multi layer ceramics is added to prevent fission products diffusion through the clad. GFR fuel development and design studies are presented in J. Somers. [3]

Core design and performance:

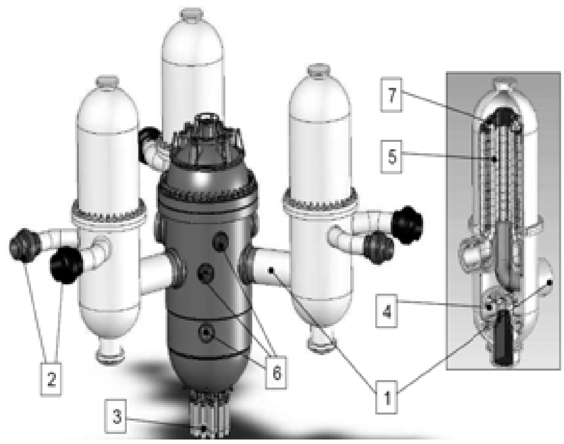
The core layout (246 fissile subassemblies, 24 control rods) has been chosen to be consistent with the maximum power derived from thermo-mechanical and thermal-hydraulic analyses, the requirements of the reactivity control system and the optimized power distribution. The main characteristics of a reference core are summarized in the table below.

GFR 2400 MWth, Reference core	
CORE – SUB-ASSEMBLY	
H/D fissile core	0.62
Inter-assembly gap (mm)	3
Fissile height (mm)	2349
He blade thickness between two plates (mm)	4.00
FUEL ELEMENT	
Plate thickness (mm)	8.4
Clad thickness (mm)	0.85
Internal liner (µm)	40+10 = 50
Pellet diameter (mm)	11.285
Pellet height (mm)	6.5
OPERATING CONDITIONS	
Core pressure drop (MPa)	0.14
Tmax fuel (°C)	1318
Tmax clad (°C)	920
CERAMIC PLATE CORE – MAIN FEATURES	
TRU enrichment (%)	18.2
Core management (eq. full power days)	3×600=1800
Average discharge burn up (at% FIMA)	6.7
Breeding Gain	-0.03

Primary system:

The reactor pressure vessel is a large metallic structure (inner diameter 7.3 m, overall height 20 m, weight about 1000 tons, and thickness of 20 cm in the belt line region).

The material selected, a martensitic 9Cr1Mo steel (industrial grade T91, containing 9% by mass chromium, and 1% by mass molybdenum) undergoes negligible creep at operating temperature (400°C). The reference material for the internals is either 9Cr1Mo or stainless steel, typically SS316LN. The global primary arrangement is based on three main loops (3×800 MWth), each fitted with one IHX–blower unit, enclosed in a single vessel.



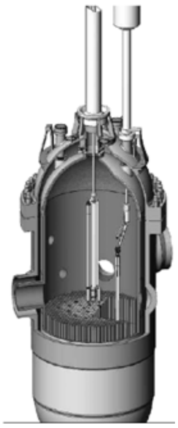
GFR primary system:

1. *Primary cross-duct*
2. *Secondary pipes with isolating valves*
3. *Control Rod Drive Mechanisms*
4. *Primary blower and associated motor*
5. *Compact Heat Exchanger modules*
6. *Pipe connections for Decay Heat Removal systems*
7. *Primary isolation valve*

This component limits the consequence of a concomitant first and second safety barriers rupture (the fuel clad and the primary system).

Specific loops for decay heat removal in case of emergency are directly connected to the primary circuit using a cross duct piping, in extension of the pressure vessel, and are equipped with heat exchangers and forced convection devices.

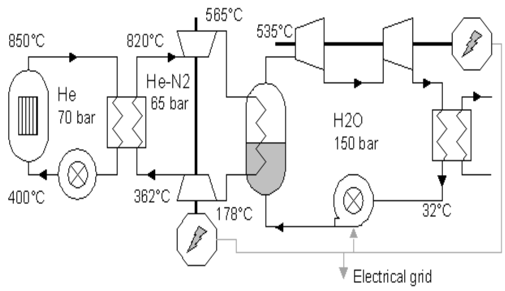
This system arrangement allows the residual power to be extracted in any accidental situations. In addition, thanks to the low pressure drop of the core design, a passive gas natural circulation can be used in most of the situations.

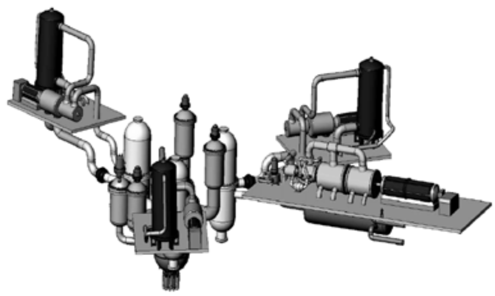


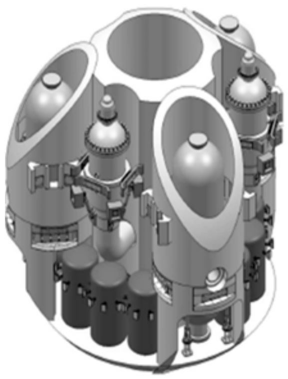
The fuel handling system is based on a jointed arm system, with fuel element loading and unloading using a fuel storage drum *via* lock chambers, the vessel being closed, as shown in the figure on the right. A dedicated forced convection device, located outside the reactor vessel, is designed to cool the spent fuel sub-assembly during its handling.

Power conversion system:

The current choice is the indirect combined cycle with He-N_2 mixture for the intermediate gas cycle. The cycle efficiency is approximately 45%, based on assumed component efficiencies and pressure drops. A schematic view of this power conversion system is shown below.





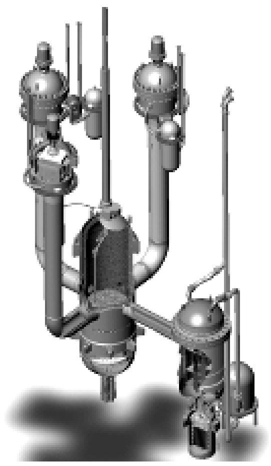


A gas tight envelope, acting as additional guard containment, has been designed to provide and maintain a backup pres-sure in case of large gas leak from the primary system. It is a metallic structure, initially filled with nitrogen slightly over the atmospheric pressure to reduce air ingress capabilities.

IV. TOWARDS A DEMONSTRATION REACTOR

Finally, an experimental demonstration and technology reactor, named ALLEGRO, is proposed to be built in the coming decades.

With a thermal power around 80 MWth, it will not produce any electricity. At first, it is foreseen to demonstrate the viability of the GFRs system file, no reactor of this type having been ever built before. ALLEGRO incorporates, at a reduced scale, all the architecture and the main materials and components foreseen for the GFR, not included the power conversion system. Its safety principles are those proposed for the GFRs: core cooling through a gas circulation in all situations, ensuring a minimal pressure level in case of a leak thanks to a specific guard containment surrounding the primary system. It will also mainly contribute to the development and qualification of an innovative refractory fuel element that withstands high temperature levels, which is one of the key points to assess the GFR system.



V. GFR AS A PLATFORM FOR EDUCATION

A number of PhD and Master Student studies related to detailed analysis of different aspects of GFR transient and steady-state neutronics, thermal-hydraulics and fuel behavior have been performed or underway. Examples of PhD studies are:

- Development and application of an advanced fuel model for the safety analysis of the Generation IV GFR. [9] A new fast-running 2D computer model of the plate-type GFR fuel was developed, benchmarked against 3D finite-element simulation and applied to the safety analysis, providing thus considerable improvement in the fuel temperature predictions in accidental situations.
- Development of the control assembly pattern and dynamic analysis of the Generation IV GFR. [10] The work has contributed to the detailed elaboration of the GFR control assembly system, including neutronics and thermal-physical aspects. The comprehensive 3D analysis of control rod withdrawal accidents has provided better understanding of the dynamic response of the GFR core to asymmetric reactivity perturbations.
- Improvement of the inherent and passive safety characteristics of the GFR. [11] A number of improvements of the GFR DHR capability under accident conditions were analytically studied, including the use of the gas-gas DHR heat exchanger, heavy gas injection in loss-of-coolant accident, use of Brayton cycle for DHR, etc.

VI. CONCLUSION

The GFR system is a Helium-cooled fast neutron spectrum reactor. It takes advantage of the fast spectrum for long term resources sustainability, in terms of use of uranium and waste minimization, and of the high temperature gas coolant for high thermal cycle efficiency and industrial use of the generated heat.

A coherent development of all the components (fuel element, core, primary system, large components) has been done through the GIF collaboration together with evaluation of safety and performance that bring today a positive image of such a technology.

A development program has been set-up among the countries that contribute to its study that should validate the fuel and technology options by 2020.

Acknowledgements

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Nomenclature

DHR	Decay Heat Removal
EFR	European Fast Reactor
FIMA	Fission of Initial Metal Atoms
GIF	Generation IV International Forum
GFR	Gas-cooled Fast Reactor
HTR	High Temperature Reactor
RPV	Reactor Pressure Vessel
SRP	System Research Plan
TRU	Trans-Uranium element

References

1. US-DOE Energy Research Advisory Committee, Generation IV International Forum, "A Technology Roadmap for Generation IV Nuclear Energy Systems," Report GIF-002-00, December 2002.
2. GIF, Provisional System Steering Committee, System Research Plan for the Gas-cooled Fast Reactor (GFR), (2008).
3. Somers, J., *et al.*, "The Generation IV GFR Fuel and Other Core Material Project", Proceedings of the GIF Symposium 2009, Paris, France, September 9-10, (2009).
4. Malo, J.Y., *et al*., "GFR 2400MW. Status of the conceptual design studies and preliminary analysis", ICAPP'09 proceedings, Japan (2009).
5. Chauvin, N., *et al.,* "GFR fuel and core pre-conceptual design", GLOBAL 2007 proceedings, USA (2007).
6. Poette, C., *et al.*, "GFR demonstrator ALLEGRO design status", ICAPP'09 proceedings, Japan (2009).
7. Richard, P., *et al.*, "Status of the pre-design studies of the GFR core", PHYSOR 2008 proceedings, Switzerland (2008).
8. Bertrand, F., *et al.*, "Preliminary safety analysis of the 2 400 MW GFR", ICAPP'08, USA (2008).

9. Petr Petkevich, "Development and application of an advanced fuel model for the safety analysis of the Generation IV gas-cooled fast reactor". EPFL thesis, No. 4180 (2008) http://library.epfl.ch/theses/?nr=4180.

10. Girardin, G., G. Rimpault, F. Morin, J.C. Bosq, P. Coddington, K. Mikityuk and R. Chawla, "Development and characterization of the control assembly system for the large 2 400 MWth Generation IV gas-cooled fast reactor", Annals of Nuclear Energy, Vol. 35, Issue 12, December 2008, Pages 2206-2218 (2008).

11. Epiney, A., K. Mikityuk, R. Chawla. "Gas-cooled fast reactor: studies on safety enhancement", Proceedings of 3rd ENEN PhD Event 2009, International Youth Conference on Energetics, Budapest, Hungary, (2009).

THE GENERATION IV PROJECT "GFR FUEL AND OTHER CORE MATERIALS"

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I. INTRODUCTION

The GFR combines the advantages of a fast neutron spectrum with those of high temperatures. [1] It can be deployed for closed fuel cycles for the minimisation of wastes, when the minor actinides are recycled. Furthermore, the effective utilisation of uranium resources is increased dramatically compared with today's light water reactor (LWR) fleets. For sustainability, self generation of Pu in the core is required, and can be provided by the fast neutron spectrum, a high power density (ca 100 $MW.m^{-3}$) and dense fuels. Proliferation risk is minimised through the use of a core without blankets. The high outlet temperatures potentially provide improved economy in the power conversion units and also permit process heat applications, just as for its thermal counterpart the very high temperature reactor (VHTR). Indeed, it is possible to gain synergy with VHTR projects, as many system components outside the core will be common to both reactor types.

Inside the core, however, there will be little similarity between the VHTR and GFR. The fuel concept for the VHTR requires coated particle fuel, which is distributed in a graphite matrix. The consequent low power density, and thermalised neutron spectrum would be inconsistent with the needs of the GFR. Initially coated particle designs (larger kernels, thinner layers) were considered for the GFR, but are no longer pursued.

Given the high core outlet temperatures exceeding 800°C, it is clear that conventional metallic core structures and fuel cladding will be unable to meet the demanding requirements, under both normal and off-normal operating conditions. Thus, only refractory metals or ceramic components can be considered for these purposes, as temperature excursions above 1 600°C have to be foreseen. Core preservation during severe accidents is a necessity, and safety considerations require that there is a limited fission gas release during transients.

The road to the first GFR demonstration reactor foresees the implementation of a test reactor, ALLEGRO, with a power of about 50 MW. For this reactor two cores are foreseen

The startup core will operate at lower outlet temperatures than planned for the GFR, and will incorporate systems to ensure its safe operation, while utilising steel cladding materials and structures. Standard MOX fuel is also considered. This initial core will also be fitted with experimental locations where ceramic fuel (plate or pin) sub assemblies can be inserted and tested.

The second core will then be fully ceramic with an advanced fuel, possibly mixed metal carbide (MC).

The design and development of an innovative refractory fuel in an advanced ceramic cladding remains a fundamental goal of the GFR system. Today, the focus is on SiC composite as structural and cladding material, with carbide (MC) fuel taking first priority over both oxide and nitride fuels as backups.

II. FUEL AND FUEL ELEMENT DESIGN OPTIONS

The classical fast reactor fuel concept consists of a fuel pin into which fuel pellets are loaded. Key design parameters are the pellet cladding gap (necessary to allow fuel swelling), and its concomitant rather poor thermal conductivity due to the helium gas bond. A plenum is available and dimensioned to collect the fission gas (and helium if minor actinide (MA) bearing fuels should be deployed). It is a goal of this Generation IV project to test new and radically innovative concepts, such as the plate type, where the fuel is in the form of a disk (low height to diameter ratio). Initially monolithic SiC was considered as a cladding material, but was dismissed in favour of composite materials such as fibre reinforced SiC, denoted SiC-SiC_f, which exhibits superior mechanical properties. Substantial efforts are needed not only in the design of the fuel and its cladding, but also in the full fuel sub assembly, necessitating a multitude of studies on various fuel / cladding configurations.

The situation is in fact far more complex, as compatibility tests have shown that SiC and mixed metal carbide fuels (MC) react, neces–sitating the introduction of a protective liner made of W or other refractory material. This W liner actually has a dual role, namely to inhibit fuel clad interaction, and to act as a fission product barrier. Thus it must be sealed at the fabrication stage. The surrounding SiC-SiC_f cladding then provides the mechanical support.

II.1 Plate Type Fuel

The basic design of plate type fuel is based on two ceramic plates, which enclose a honeycomb structure containing cylindrical pellets made of the mixed carbide fuel. This design is shown schematically in Figure 1. The individual plates can then be stacked in a fuel assembly as shown in Figure 2. [2]

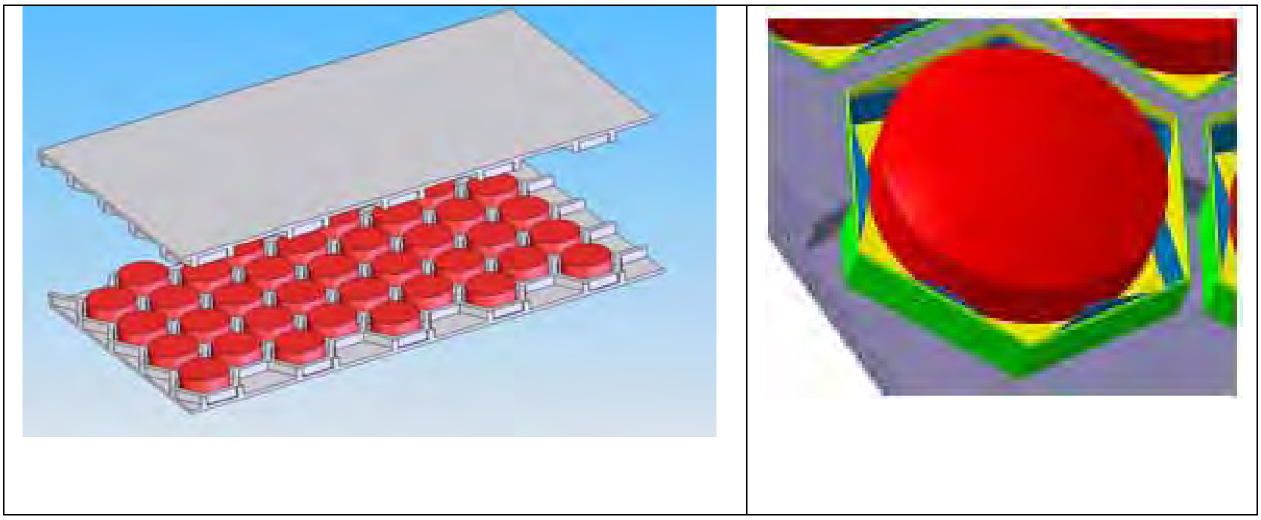

Figure 1: GFR plate fuel design

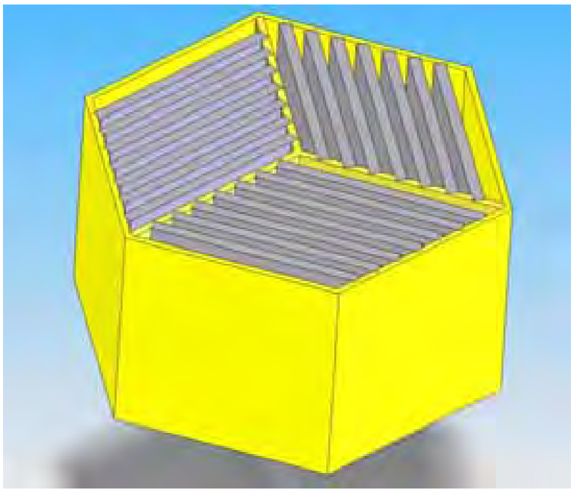

Figure 2: Plate fuel sub assembly concept

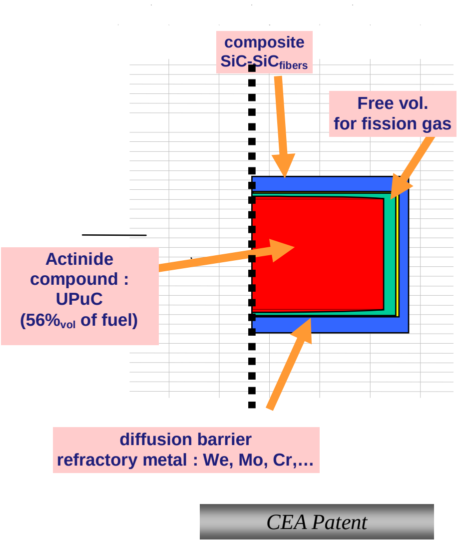

Figure 3: Cell dimensions in plate type fuel

The design of the plate fuel concept has been determined in a set of detailed thermo-mechanical assessments made at the CEA. Parameters investigated include the cell dimensions, fuel disk geometry, radial and axial gaps, and free volume along with irradiation behaviour laws derived for (U,Pu)C pellets and SiC-SiC_f cladding (see Table 1 and Figure 3).

A particularly important design parameter is the closed axial gap between the pellets and SiC-SiC_f cladding at begin of life (BOL), which provides a mechanical bond decreasing the thermal barrier and consequently the fuel operating temperature. The calculated thermal distribution at BOL (1/24 cell) is shown in Figure 4. The hottest part of the fuel is at about 1500 K and remains constant throughout the irradiation. The temperature gradient in the cladding material is higher than in the fuel itself. This is a consequence of the poor thermal conductivity of the SiC-SiC_f cladding, and clearly, a solution to ameliorate this situation is required. The high temperature gradient in the cladding also results in inhomogeneous thermal expansion, and combines with pellet-clad interaction, to give a compressive stress in the cladding of 300 MPa at the so called P12 point. [2]

Table 1: Plate Fuel Cell Parameters

Fuel plate thickness	ep =8.4 mm
Across flat cell	a = 14 mm
Cladding thickness	eg = 0.85 mm
Pellet-clad axial gap	ja = 0.05 mm
Pellet-clad radial gap	jr = 0.75 mm
Wall thick in honey comb	ev = 1.115 mm
Pellet diameter	D = 11.285 mm
Pellet height	hc = 6.5 mm
Fissile vol by cell	23%

The much smaller thermal gradient in the fuel limits thermally induced stresses. It is assumed that fuel swelling will be accommodated by creep, which due to the operating pressure on the plate (70 bar) should occur in the radial direction and not cause any additional axial stress on the cladding. The thermomechanical calculations are encouraging, but it is clear that further evaluations are needed, and in particular both integral and separate effect irradiation tests are required to prove the feasibility of the concept. Fabrication of such a complex structure with individually sealed alveoli will be a key issue to be mastered.

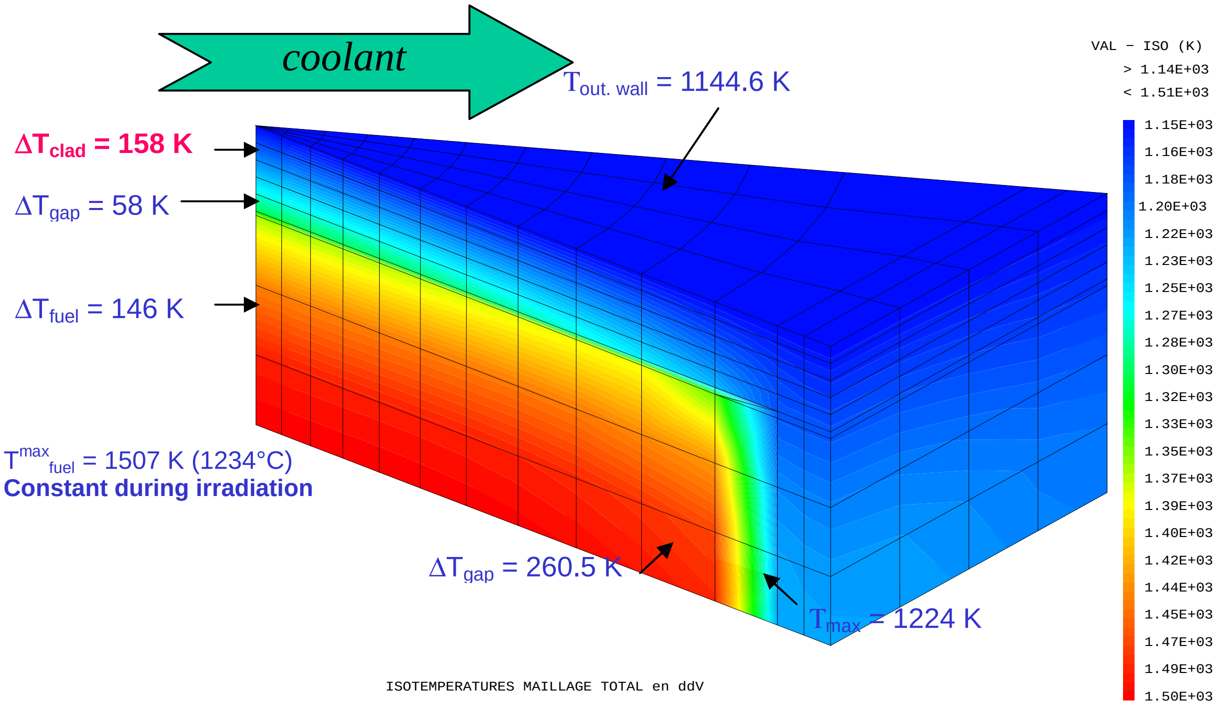

Figure 4: Thermal behaviour of a plate type fuel cell at BOL. Temperature gradients in the cladding, axial and radial gaps, and fuel are shown

II.2 Pin type fuel

Pin type fuel will be similar to current fast reactor designs. A single rod should be formed from two individual components, with independent seals and gas plenums. This improves mechanical stability, and it is believed that manufacturing of short straight tubes will be simpler than a single longer component. Again carbide fuel (with oxide and nitride as back up options) and SiC-SiC$_f$ cladding are considered. Irradiation induced carbide fuel swelling (twice that of the oxide) can cause fuel clad mechanical interaction (FCMI) at relatively low burnups. Slender pins, certainly not greater in inner diameter than 5 mm, are required and the pellet cladding gap should be chosen carefully to allow for fuel swelling and deleterious mechanical interaction. Spherepac fuel would also be advantageous, as it provides a convenient means to accommodate swelling, while limiting stresses on the cladding. Pellet fuel at 85% of the theoretical density can also accommodate swelling internally within its porous structure. A high thermal stability is required, however, to avoid sintering at BOL, which would then result in higher fuel temperatures due to the concomitant increased gap size. The studies on pin type fuel are still at an early stage. Though there are difficulties to master *vis-à-vis* fuel swelling and concomitant FCMI that could ensue, manufacturing of the components and sealing them should be simpler than for plate type fuel.

III. MATERIALS

III.1 Fuel

For neutronic and safety design studies, carbide fuel has advantages over oxide fuel. It has a high fissile element density, and also higher thermal conductivity. In principle, nitride fuel has similar properties, but has the disadvantage of requiring ^{15}N enrichment, to avoid ^{14}C produc–tion during irradiation. Past experience has shown that its fabrication may be somewhat more convenient than carbide, and consequently, it is considered as a reserve option along with the oxide. Some properties of various fuel compositions are summarised in Table 2. A drawback of carbide fuel is its volatility, in particular if MA bearing fuels are considered. The EURATOM FP6 programme GCFR [3] has produced a number of review reports covering fabrication properties, past irradiation programmes and reprocessing of MC and MN fuels. The CEA, ITU and PSI have been actively engaged in the re-establishing fabrication facilities for such advanced fuels. Mainly carbothermal reduction of the oxides is considered for the production of nitride and carbide fuels. It is proposed that actinide losses during production (due to the high vapour pressure of Pu) can be overcome, if a solid solution of (U, Pu)O_2 is used as the starting material. As the solid solution is already formed, the losses should be lower as PuC should not be formed, rather (U, Pu)C directly. In addition, the length of the high temperature processing in the carbothermal reduction step will be shorter. Plans are also afoot at PSI to use such a process for particle production, enabling Spherepac deployment.

Table 2: Fuel Properties

	Carbide	Nitride	Oxide
Theoretical density ($g.cm^{-3}$)	13.58	14.32	11.5
Melting point (°C)	2420	2780	2750
Thermal Conductivity ($W.m^{-1}K^{-1}$)	16.5	14.3	2.9

III.2 SiC-SiC$_f$ cladding

Much of the pioneering work on advanced SiC-SiC$_f$ has been made at Kyoto University. This composite material has key advantages in terms of toughness (K_{IC} up to 30 $MPa.m^{1/2}$). Its permeability could necessitate an outer liner to prevent damage of the inner liner by the helium at 70 bar in the primary loop of the reactor. There also remain some concerns about the stability of SiC at very high temperatures that could be encountered during a severe temperature excursion in off normal operating conditions

Several irradiation experiments on monolithic SiC were performed in the UK in the 1960s. The results and experience were gathered together in reports produced within the EURATOM FP6 programme GCFR. [1] For other reactor structural components, *e.g.* the reactor pressure vessel and reflector, synergy is sought with the Generation IV VHTR project on structural materials.

IV. IRRADIATION TESTING PROGRAMMES

Ultimately, all fuel and structural materials need to be tested and qualified in dedicated irradiation experiments. Carbide fuels were tested in various programmes in the past, but their behaviour is not nearly as well known as for oxide fuel. When MAs are present, there is no information at all. Ongoing and planned irradiation programmes are depicted in Figure 5. [4] Post Irradiation Examination on the NIMPHE programmes is in progress at JRC-ITU. These tests showed a relatively good behaviour of both carbide and nitride fuel at intermediate burnups. Recently, the results of an irradiation test on spherepac fuel made in the Fast Flux Test Facility (FFTF) [5] showed excellent behaviour of carbide fuel in this form.

New irradiation tests on carbide and nitride fuel compatibility with SiC-SiC$_f$ are nearing completion in Phenix (FUTURIX CONCEPT), while a new programme in BR2 (IRRDEMO) will be launched shortly. Material tests are ongoing in Phenix and OSIRIS (FUTURIX MI and REA series, respectively).

V. THE FUEL AND OTHER CORE MATERIALS PROJECT

The FCM project arrangement will be signed by France, Switzerland, Japan and EURATOM. The programme consists of the following work packages:

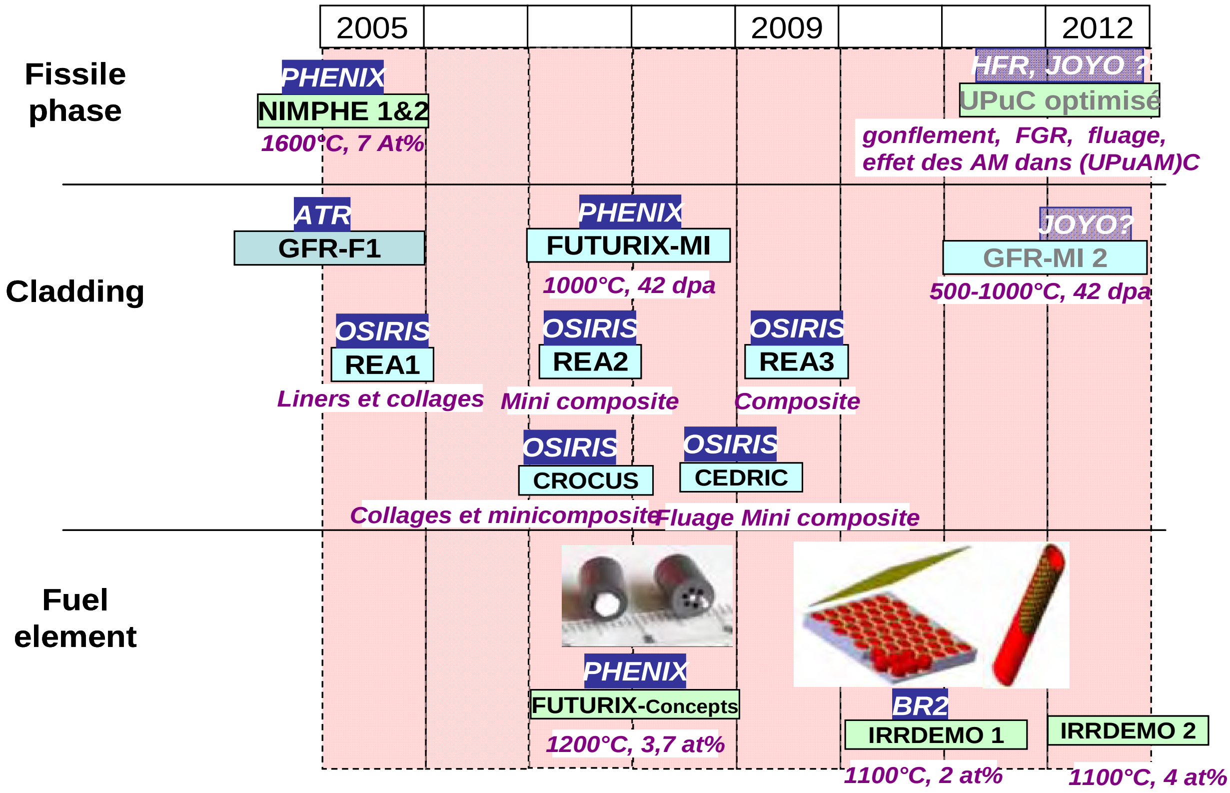

Figure 5: Ongoing and planned irradiation experiments on GFR fuel and materials

1. **Fuels and assemblies modelling and design**, which covers studies on fuels in plate and pin geometries, using fuel performance codes (PLEIADES-CELAENO, TRANSURANUS) and finite element methods. Thermo aerodynamic codes are used for assembly design. Fuels under consideration include MC, MN and MOX, while cladding and structural materials include $SiC\text{-}SiC_f$ and oxide dispersion strengthened steel (ODS), the latter being an option for the ALLEGRO start up core.

2. **Basic fuel material studies**, covers irradiation studies on inert materials in Phenix, detailed investigations on fresh and irradiated MC and MN fuels, and interactions between actinide fuels, fission products and inert cladding material.

3. **Basic in core material studies** is dedicated to the structure materials for the core (subassembly, control rods, guide tubes, and reflector) and in particular their ability to satisfy main safety requirements of the reactor in various operating conditions. Out of pile and in pile studies are foreseen, along with the development of appropriate codes and standards. Where possible synergies with the Generation IV VHTR Materials project are sought.

4. **Fuel fabrication process development** covers the comparison and selection of processes suitable for MX fuels with and without minor actinides, testing of these processes with the development of relevant flowsheets, leading to the fabrication of U, U-Pu and U-Pu-MA specimens for property determination and eventually for irradiation testing.

5. **Fuel and assembly development and qualification by irradiation testing** addresses screening, optimisation and validation phases required for fuel deployment. Minor actinide bearing fuels are considered in a first instance in dedicated separate effect studies, possibly

in material testing reactors, before validation in fast reactor systems.

6. **Behavior during off-normal conditions** is concerned with the ability of the fuel and fuel element to retain FPs during a depressurisation accident. This activity covers thermal behaviour of SiC-SiC$_f$ in appropriate conditions, high temperature behaviour of fresh MX fuels, chemical compatibility of all core materials at high temperature, and the tightness of the fuel element at temperatures beyond 1 600°C (corresponding to severe accidents).

The GFR FCM project will provide design data for ALLEGRO and GFR cores. It relies heavily on the French national programme, while Japanese programmes contribute mainly to SiC-SiC$_f$ investigations, and Switzerland and EURATOM contribute mostly on fuel issues.

VI. CONCLUSION

The GFR fuel and other core materials (FCM) project arrangement awaits signature, but even without formal signature the partners have collaborated effectively over the past 5 years. The project programme has now been defined until 2012. At that time decisions on future systems will be made in France and also at the European level in the framework of the Sustainable Nuclear Energy Technology Platform (SNE TP). [6]

References

1. Malo, J.Y., T. Mizuno, T.Y.C. Wei, C. Mitchell, P. Coddington, GIF GFR end of exploratory phase design and safety Studies, Proceedings of International conference on the Physics of Reactors "Nuclear power A sustainable Resource", Interlaken, September 2008.
2. Chauvin, N., Y. Malo, J.C. Garnier, F. Bertrand, J.C. Bosq, A. Ravenet, D. Lorenzo, M. Pelletier, J.M. Escleine; I. Munoz, J.M. Bonnerot, GFR Fuel and core preconceptual design studies, Proceedings of the Global 2007 Conference, Idaho, September 2007.
3. www.gcfr.org.
4. Basini, V., International Seminar of GFR, Paris, February 2008.
5. Bart, G., F.B. Botta, C.W. Hoth, G. Ledergerber, R.E. Mason, R.W. Stratton, AC3-irradiation test of spherepac fuel in the US Fast Flux Test Facility, J. Nucl. Mater., 376(2008)47.
6. www.snetp.eu.

SCWR: OVERVIEW

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I. INTRODUCTION

The Supercritical Water-Cooled Reactor (SCWR) is a high-temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point of water (above 374°C, 22.1 MPa). The main advantage of the SCWR is improved economics because of the higher thermodynamic efficiency and plant simplification opportunities that are made possible by the use of a high-temperature, single-phase coolant. However, there are opportunities to achieve improvements in other areas such as safety, sustainability, proliferation resistance and physical protection because of the flexibility of the various design options that include fast and thermal spectra as well as opportunities to utilize conventional or advanced fuel and fuel cycles.

The SCWR is the only water-cooled Generation IV reactor concept and builds on many years of experience in advanced water-cooled reactor and supercritical fossil plant development. It also builds on proven advanced concepts and systems from both industries (*e.g* turbine technology). The main challenge in the SCWR development is to combine advanced reactor technology with supercritical fossil technology so that the desired operating conditions are produced by nuclear heating rather that fossil fuel. This introduces challenges in the selection of materials for the core components that will require significant R&D. In addition R&D will be needed in other areas such as thermal-hydraulics to produce data and information needed to design and license the reactor. A system research plan has been developed by the GIF SCWR System Steering Committee that outlines the R&D requirements for the SCWR development[1]. The GIF members that are currently active in the SCWR R&D include: Canada, EURATOM, France, Japan, Republic of Korea, and China (as observer).

II. SCWR DESIGN OPTIONS

The SCWR can be designed as a fast or thermal reactor with closed or once-through fuel cycle. In addition, pressure-vessel or pressure-tube designs can be used which opens the way for a number of design options that have the potential to significantly improve the four GIF metrics.* A schematic of the SCWR is shown in Figure 1. Table I lists the main operating parameters and features of the SCWR.

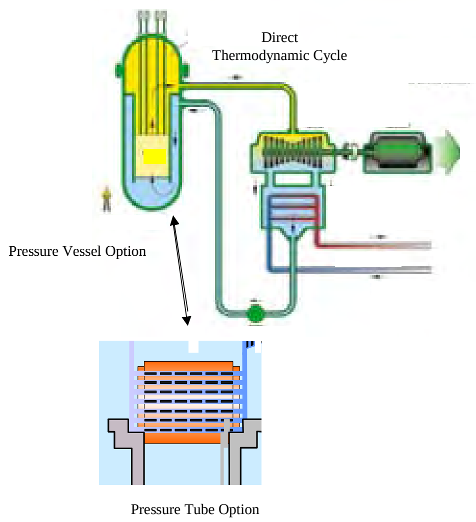

Figure 1: SCWR Schematic

* Economics, safety, sustainability, proliferation resistance and physical protection (PRPP)

TABLE I: SCWR Reference Parameters

Parameter	Reference value(s)
Power (MWe)	Up to 1500
Pressure (MPa)	25
Inlet Temperature (^oC)	Up to 350
Outlet Temperature (^oC)	Up to 625
Efficiency	Up to 50%
Burnup (thermal option)	Up to 60GWd/tHM
Burnup (fast option)	Up to 120GWd/tHM
Spectrum	Thermal or Fast
Fuel	UO_2, MOX, thorium
Fuel Cycle	Once through or Open
Pressure Boundary	Pressure tubes or pressure vessel
Coolant	Light water
Moderator	Light water or ZrH_2 (PV) or heavy water (PT)

The SCWR GIF members are considering several design options that are based on the parameters in Table I. These design options include:

1. University of Tokyo thermal and fast spectrum designs:[2] these are pressure-vessel design concepts (see Figures 2, 3) that have been under development at the University of Tokyo since 1989 (thermal version) and 2005 (fast version). The thermal version is called “Super LWR” and the fast reactor version is called “Super Fast LWR”.
2. High Performance Light Water Reactor (HPLWR):[3] this is a pressure vessel design that is under development in Europe and is partially funded by the European Commission (see Figure 4).
3. CANDU®†-SCWR:[4] this is a pressure-tube reactor that is being developed by AECL that uses a thorium fuel cycle and a separate heavy water moderator with enhanced safety functions (Figure 5).
4. SCWR-SM:[5] this is a pressure vessel design under development in the Republic of Korea that utilizes a solid ZrH_2 moderator (Figure 6).
5. Mixed core design:[6] this is a pressure vessel concept that is being evaluated at Shanghai Jiao Tong University. The core consists of a fast spectrum inner region and a thermal spectrum outer region (Figure 7).

The above design options have common features such as the use of a direct thermodynamic cycle that contributes to plant simplification and improves efficiency. They also have common R&D needs that will be described later. These different designs are expected to result in very economical reactors with safety features that are at least equivalent to the high safety standards implemented in Generation III+ water-cooled reactors with opportunities for further enhancements.[7,8,9] In addition, the introduction of the fast spectrum and the use of advanced fuel cycles[10,11] provide opportunities for further enhancements in sustainability and proliferation resistance and physical protection. These designs are being developed with the objective of providing options and ideas that can potentially be used to design an advanced water-cooled reactor that enhances all GIF metrics. These design activities are used to guide the R&D activities that are described next.

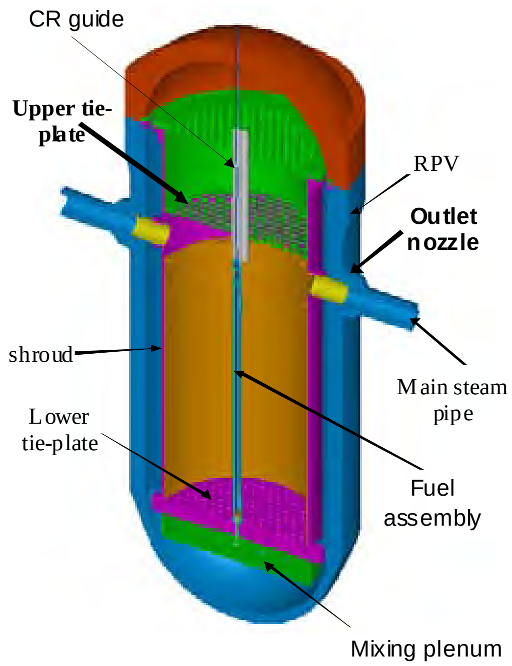

Figure 2: Super LWR Schematic.

† CANDU – Canada Deuterium Uranium, a registered trademark of Atomic Energy of Canada Limited (AECL).

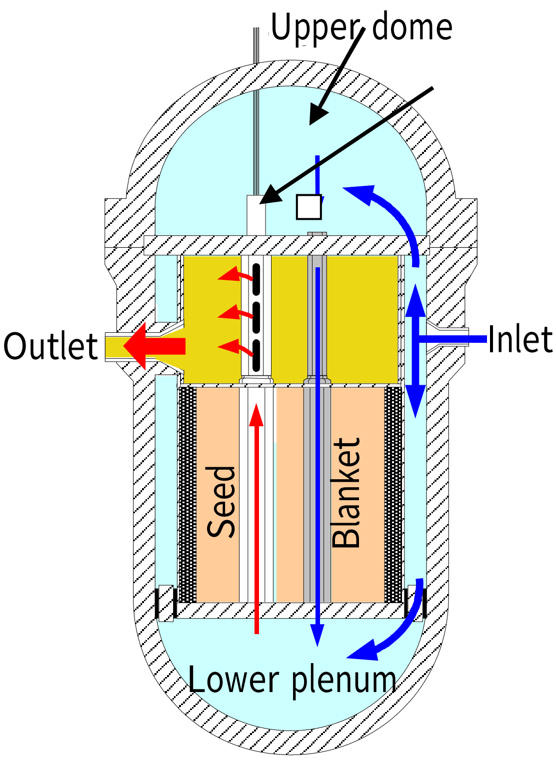

Figure 3: Super Fast LWR Schematic.

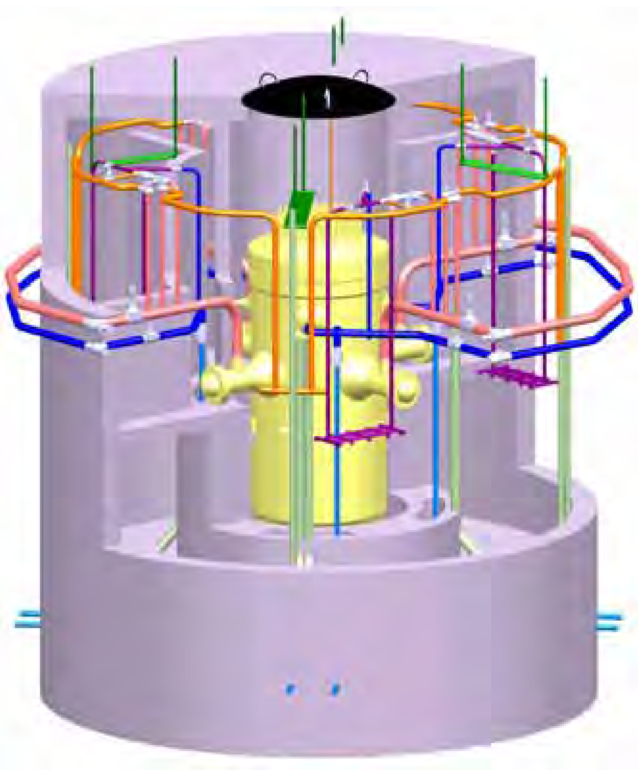

Figure 4: HPLWR Schematic.

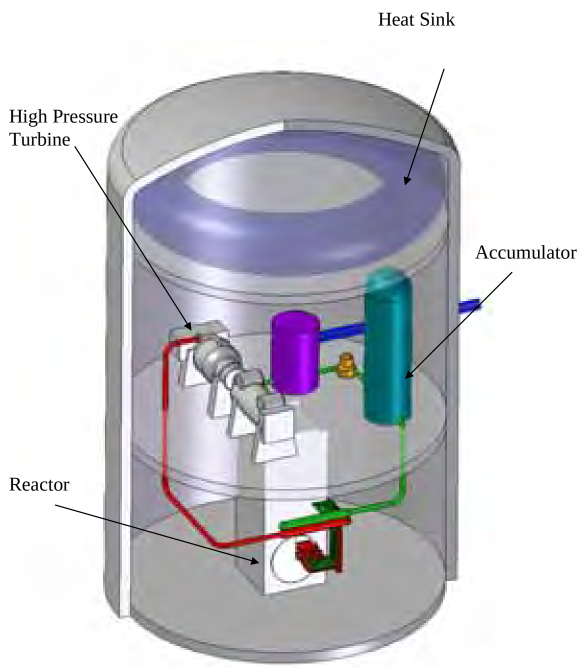

Figure 5: CANDU-SCWR Schematic.

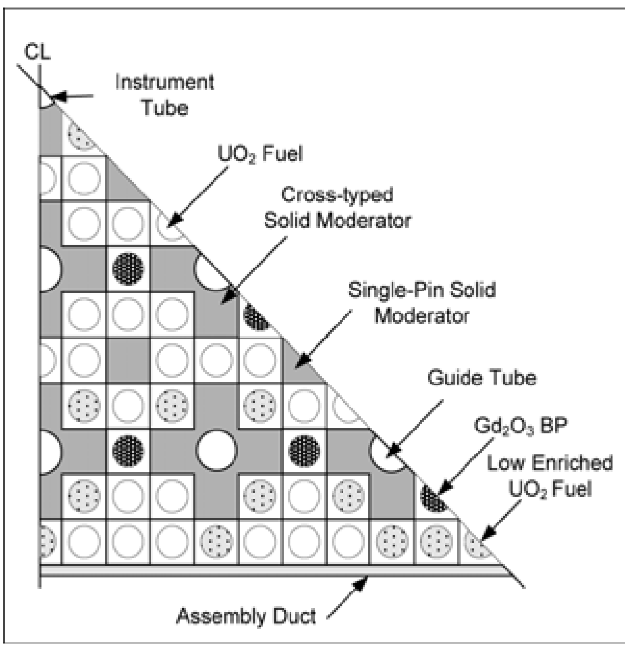

Figure 6: SCWR-SM Fuel Assembly with a Solid Moderator.

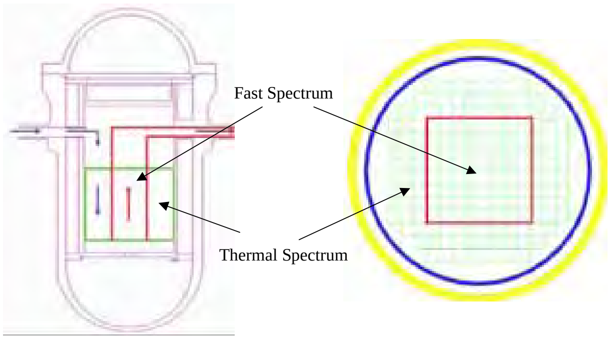

Figure 7: Mixed Spectrum Concept.

III. SCWR R&D

The collaborative GIF R&D projects focus on areas that are common to the design options under considerations by the SCWR members. Two major collaborative R&D projects are currently underway:

1. Materials and chemistry: this project involves testing of key materials for use both in-core and out-core, for both the pressure tube and pressure vessel designs. A reference water chemistry will also be investigated, based in large part on materials compatibility and the radiolysis behavior.

2. Basic thermal-hydraulic phenomena, safety, stability and methods development: this project will address knowledge gaps that exist in key areas such as heat transfer and critical flow at supercritical conditions. The design-basis accidents for an SCWR will have some similarities with conventional water reactors, but the different thermalhydraulic behavior and large changes in properties around the critical point compared to water at lower temperatures and pressures will have to be better understood.

More details of these collaborative programs will be presented at this symposium.

Together with the design activities, the above R&D areas are on the critical path and are needed to establish the viability of the SCWR in meeting GIF goals and objectives. Other R&D areas that are underway for specific designs include advanced fuel and fuel cycles (*e.g.*, using thorium in the pressure-tube design and the development of the fast core and mixed core options for the pressure vessel design), and hydrogen production.

The materials and thermal-hydraulics R&D projects will utilize test facilities to perform basic tests to provide information needed to optimize the various designs. In addition, major test facilities to qualify certain aspects of the SCWR (*e.g.*, fuel qualification) have been identified and collaborative projects have been initiated to design and build these facilities.

The SCWR R&D is expected to provide sufficient information by ~2020 to enable the design, licensing and construction of a prototype.

IV. CONCLUSION

The SCWR is a water-cooled reactor that operates above the thermodynamic critical point. Several design options using pressure vessel and pressure tube technologies are currently under consideration with the aim of providing a spectrum of possibilities for consideration for the next generation of water-cooled reactor technology. These design options are being used to define high-priority R&D areas and will contribute to the definition of a future design that will improve and optimize all GIF metrics.

Acknowledgements

The author acknowledges the input and review from the SCWR Steering Committee members. Financial support for this work was provided by NRCan.

Nomenclature

AECL	Atomic Energy of Canada Limited
CANDU	Canadian Deuterium Uranium Reactor
CR	Control Rod
RPV	Reactor Pressure Vessel
GIF	Generation IV International Forum
HPLWR	High Performance Light Water Reactor
LWR	Light Water Reactor
NRCan	Natural Resources Canada
PRPP	Proliferation Resistance and Physical Protection
SCWR	Super-Critical Water-Cooled Reactor
SCWR-SM	SCWR design under development in Korea
SM	Solid Moderator

References

1. Generation IV International Forum SCWR System Steering Committee, "Generation IV Nuclear Energy Systems System Research Plan for the Supercritical Water-cooled Reactor," Rev. 0 (2007).
2. Oka, Y., *et al*., "Research and development of Super LWR and Super Fast Reactor," *Proc. 3rd Int. Symp. on SCWR, Shanghai, China,* Paper No. SCR2007-I003 (2007).
3. Schulenberg, T. and J. Starflinger, "European Research Project on High Performance Light Water Reactors," *Proc. 3rd Int. Symp. on SCWR, Shanghai, China*, Paper No. SCR2007-I001 (2007).
4. Khartabil, H.F., *et al*., "The Pressure-Tube Concept of Generation IV Supercritical Water-Cooled Reactor (SCWR): Overview and Status," *Proceedings of ICAPP '05, Seoul, KOREA* (2005).
5. Bae, Y.Y., *et al*., "Update on the SCWR Research in Korea," *Proc. 4th Int. Symp. on SCWR, Heidelberg, Germany* (2008).
6. Cheng, X., *et al*., "A Mixed Core for Supercritical Water-Cooled Reactors," *Proc. 3rd Int. Symp. on SCWR, Shanghai, China*, Paper No. SCR2007-P021 (2007).
7. Khartabil, H. (2008), "Review and Status of the Gen-IV CANDU-SCWR Passive Moderator Core Cooling System," *Proc. 16th Int. Conf. on Nuclear Eng. (ICONE-16), Orlando, FL, USA* (2008).
8. Bittermann, D., *et al*., "The Safety Concept of the HPLWR," *Proc. 4th Int. Symp. on SCWR, Heidelberg, Germany* (2008).
9. Ikejiri, S., *et al*., "Research and Development of a Super Fast Reactor (6): Analyses of Abnormal Transients," *16th Pacific Basin Nuclear Conference (16PBNC), Aomori, Japan,* Paper No. P16P1295 (2008).
10. Cao, L., *et al*., "Research and Development of a Super Fast Reactor (4): Transmutation Analyses of Minor Actinides and Transuranium Elements," *16th Pacific Basin Nuclear Conference (16PBNC), Aomori, Japan,* Paper No. P16P1293 (2008).
11. Duffey, R.B., "Future Fuel Cycles: A Global Perspective," *Proceedings of the 16th International Conference on Nuclear Engineering ICONE16*, Orlando, Florida, USA (2008).

STATUS OF ONGOING RESEARCH ON SCWR THERMAL-HYDRAULICS AND SAFETY

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I. INTRODUCTION

SCWR (**S**uper**C**ritical pressure **W**ater-cooled **R**eactor) is an advanced reactor concept which has advantages in improving economics while utilizing most of the existing PWR and BWR technologies as well as fossil fuel power plant technologies. The Generation IV International Forum selected the SCWR as one of the viable candidates of a nuclear power plant to be deployed by 2030, especially for economic electricity generation. In order to organize and support the multi-lateral collaboration among the member countries, the SCWR Steering Committee has been established in March, 2007, and several Project Management Boards (PMBs) such as Thermal-Hydraulics and Safety, Material and Chemistry, System Integration and Assessment, and Fuel Qualification have followed.

Since the formation of the provisional SCWR Thermal-Hydraulics and Safety Project Management Board (TH&S PMB) in November 2004, the participating countries, Canada, EU (as a consortium), Japan (as a consortium), and Korea have been working hard on the preparation of the Project Plan (PP). This plan is a kernel part of the Project Arrangement (PA) and will serve as a basic document for the forthcoming multi-lateral collaboration. The endorsement of the PA by the participating countries is expected to be completed no later than the end of 2009. The PP describes the coordinated research activities in the technology development area (TDA) of SCWR TH&S. In the System Research Plan (SRP), the required research items have been identified and these include heat transfer, hydraulic characteristics, critical flow, identification of safety requirements and evaluation, stability, development of system codes and relevant methodologies, subchannel analysis, and simulation of system performance and behavior during transient and accident. Most of these items can be performed on an individual basis, but others may require the integrated efforts of all or some participants. The PP describes the framework of the collaboration scheme, required resources, estimated schedule and deliverables. It will be reviewed annually by the PMB members and may be modified on the suggestion of the Signatories and the approval of the System Steering Committee.

In the meantime, a considerable amount of work has been accomplished by the member countries and arrangements will be made for those identified as shared items as soon as the PA is signed. In this paper, the accomplishment of the member countries in the SCWR thermal-hydraulics and safety area is described briefly. Besides the member countries China is actively working on the SCWR thermal-hydraulics research but it will not be covered in this paper. Russia is also considering joining this PMB but its decision has not yet been made.

II. CANADIAN THERMAL-HYDRAULICS AND SAFETY PROGRAM

Thermal-hydraulics characteristics at super-critical water-flow conditions are required in support of the design and qualification of the fuel bundle and safety analyses for the SCWR. GIF participants in the SCWR development are preparing a Project Plan for thermal-hydraulics and safety research work. The plan lists tasks required for completing the conceptual design of the SCWR, and covers key areas such as heat transfer, critical flow, instability, development of analytical toolsets for supercritical-water applica-tions, and preliminary safety analyses. Completing these tasks will demand a large coordinated effort between research organizations and the academic community.

The Canadian contribution to various key areas of the GIF SCWR Thermal-hydraulics and Safety Project has been identified in the Project Plan. It consists of projects directly relevant to the CANDU SCWR fuel and core designs at AECL and fundamental research and develop-ment (R&D) projects related to the SCW flow and heat transfer at various Canadian universities.[1] In addition, AECL has initiated other col-laborative projects with Canadian universities (with proposed support from the Ontario Research Fund) and Chinese universities to develop the future reactor design. Information from these projects is also applicable to the Generation IV SCWR design and will be included as part of the Canadian contribution to GIF. The Thermal-hydraulics and Safety projects in the grant program focus on improving / developing heat-transfer prediction methods for supercritical heat transfer in tubes and bundles, examining the stability and critical-flow characteristics of supercritical flow, and performing simulations of the depressurization phenomena through small breaks at supercritical conditions. The tube-data-based prediction method for supercritical heat transfer is applied in subchannel analyses, while the bundle-data-based prediction method is implemented for safety analyses.

The design criterion for the CANDU-SCWR is based on the cladding temperature limit for normal operation and trip analyses. Experimental data on heat transfer are crucial in establishing this limit accurately.[2] A database on supercritical heat transfer in tubes has recently been assembled.[3] It is being applied to assess various correlations and, if necessary, to improve prediction accuracy.

Figure 1 illustrates variations in the heat-transfer coefficient as a function of temperature for supercritical water flow inside a 10-mm inside diameter (ID) tube.

A project has been initiated at the University of Ottawa to develop a look-up table for heat transfer covering trans-critical conditions (*i.e.*, both the near-critical region and the supercritical region) in tubes. Advantages of the look-up table approach include superior prediction accuracy (representing directly the database), wide-ranging applicability, and a smooth transition in tabulated values between different regions.

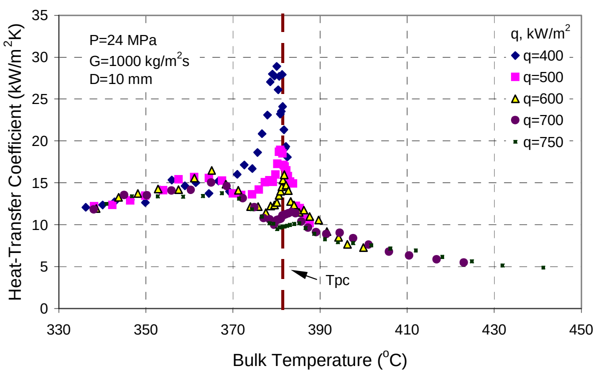

Figure 1: Heat-Transfer Coefficient in Supercritical Water Flow.

Performing heat-transfer experiments with supercritical water flow is complex and expensive due, primarily, to the harsh operating environment and the high level of required heating power. Surrogate fluids (such as carbon dioxide and refrigerants) have been suggested for replacing water in heat transfer studies. These fluids were previously utilized in studies of critical heat flux and film-boiling heat transfer at subcritical conditions. Applying these fluids reduces experimental cost and schedule, reduces test-section design and operation risk, and increases testing flexibility. This arises from the fact that supercritical conditions for surrogate fluids are less severe than those for water.

Figure 2 illustrates the range of reduced pressure and reduced temperature covered in the supercritical heat-transfer database for carbon dioxide flow.[3]

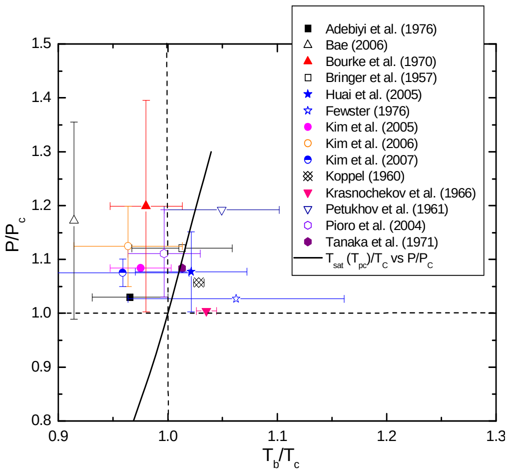

Figure 2: Range of Selected Supercritical Heat-Transfer Data for Carbon Dioxide Flow.

A project has been established to expand a sub-critical refrigerant test facility at Carleton University for supercritical heat transfer experiments. The test facility can accommodate test sections such as tubes, annuli, and small bundle subassemblies to study various separate effects on supercritical heat transfer. A tubular test section is being designed for commissioning the facility using Refrigerant-134a. The first test series examines the effect of spacing devices on supercritical heat transfer in annuli.

Large amounts of supercritical heat transfer data are available for tubes, but there is a lack of data for bundle geometries. A heat-transfer test facility has been designed for construction at the University of Ottawa (UO).[4] It employs carbon dioxide as working fluid and is capable of testing small bundle subassemblies. Key components of this test facility have been procured.

A project has been arranged to complete the construction of the facility and perform commissioning test using a tubular test section. The commissioning data will be compared against experimental data in the AECL database[3]. Another project has been established to perform heat-transfer experiments using a 3-rod bundle string in this test facility. The objective of this experiment is to generate bundle heat-transfer data in carbon dioxide flow. This data is essential for quantifying the impact of flow and enthalpy distributions in subchannels on supercritical heat transfer, and is also applicable for validating subchannel codes and computational fluid dynamic tools.

Computational fluid dynamic (CFD) tools have been widely used in support of fuel design for SCWR. These tools are based on fundamental conservation equations but depend strongly on the turbulence model selected in the calculation. Currently, there is little (or no) information on turbulence measurements in supercritical flow. A project has been awarded to the research team at the University of Ottawa to obtain turbulence measurements in a 22.9 mm tube with super–critical carbon dioxide flow. Subsequently, the measurement technique may be implemented to the 3-rod bundle.

The CANDU-SCWR may be susceptible to dynamic instability due to the sharp variation in fluid properties (such as density) in the vicinity of the critical point. This instability may lead to a high cladding temperature in the fuel, prematurely impacting on the operating and safety margins. Analytical models have been developed for predicting the onset of dynamic instability with in-phase 1D oscillations and out-of-phase 2D and 3D oscillations.[5] A project has been approved for the University of Manitoba to perform flow-stability experiments using carbon dioxide in single and parallel channels. Test data will be applied for the validation of the analytical model.

In support of the design and operation of the reactor safety (or relief) valves and the automatic depressurization system, the critical (or choked) flow characteristic must be established at supercritical conditions since current information has been obtained at subcritical conditions. This established characteristic is also required in the analysis of a postulated large-break loss-of-coolant accident event. A project has been awarded to the École Polytechnique to construct a test facility for critical-flow measurements in water at supercritical conditions. Blow-down experiments from a supercritical pressure tank to a medium pressure reservoir will be performed with discharge nozzles of different shape, size and length. Direct experimental measurements of the temperature and pressure along the discharge nozzles, and of the void fraction and flow rate at the nozzle outlet will be obtained. These data will enable accurate benchmarking of existing critical-flow models and, if needed, the development of new ones.

The basic thermal-hydraulics phenomena during hypothetical accidents involving depressurization of the reactor coolant system at subcritical water conditions (such as critical break discharge and flashing behavior) have been extensively analyzed within the nuclear industry. However, very little analysis is available for reactors near or above the supercritical pressure. A project has been initiated at McMaster University to examine depressurization characteristics for near critical and supercritical systems, taking into consideration the unique properties as the fluid transitions through the critical state. These systems include simple pipes and tank geometries (which have been previously studied at sub-critical conditions) and constricted flow passage through nozzles (simulating the small breaks phenomena). Simulation results will be compared against the experimental data at subcritical and super–critical conditions.

III. EUROPEAN CONTRIBUTION TO THE GIF THERMAL-HYDRAULICS AND SAFETY PROJECT

In Europe the High Performance Light Water Reactor (HPLWR) is currently under development. The High Performance Light Water Reactor is a Light Water Reactor (LWR) with supercritical water at 25 MPa as coolant and moderator.

A consortium of 10 partners from 8 European countries and three so-called active supporters cooperate within the "High Performance Light Water Reactor Phase 2" project which started in 2006 and will end in 2010. This project is co-funded by the European Commission. The objective of this project is to assess the feasibility of this reactor concept and to assess the economical potential of this reactor concept, see Starflinger *et al*.[6]

Most of the European research activities on the SCWR are covered by the HPLWR project in its Phase 2. The outcome of the research activities on safety and thermal hydraulics is contributing to the TH&S program within GIF. The research activities and deliverables are defined in the project plan of the TH&S project and the project plan is part of its project arrangement.

In the following two sub-sections, a brief description of the current status of the research on thermal hydraulics and the safety concept is given.

III.A. Status of research on thermal hydraulics for the HPLWR

Heat transfer from the fuel rods to super critical water is a very important research issue since it determines the temperatures of the fuel cladding. It is well known that this heat transfer is strongly influenced by large changes in the physical properties of super critical water near the pseudo critical point. The heat transfer may be enhanced or strongly decreased (called heat transfer deterioration) depending on local conditions like heat flux and flow rate of SCW. A second important aspect in the heat transfer from the HPLWR fuel rods is the wire which is wrapped around each fuel rod. The wire serves as spacer between the fuel rods and enhances mixing between the sub-channels of the fuel bundle.

As a starting point of the thermal hydraulic research work, a data base of experimental data for supercritical water in smooth heated tubes has been prepared by Loewenberg *et al*.[7] The data cover a broad range of experimental conditions for sub- and supercritical pressures. The data base has been used for the validation of CFD models that have been developed by Zhu and Laurien[8] and by Visser *et al*.[9] Besides supercritical water, also super critical CO_2 has been used for CFD code validation, showing similar physical effects.

Analytical work as well as CFD work has been performed by Palko and Anglart[10] to improve the basic understanding of heat transfer mechanisms for super critical water with a focus on the mechanisms describing the onset of heat transfer deterioration. A heat transfer mechanism identification and ranking table has been made by Anglart.[11] CFD is also used to quantify the effect of the wire wrap in representative sub-channel geometries. Recent results have been published by Himmel *et al*,[12] Chandra *et al*,[13] Laurien *et al*,[14] and Kiss *et al*.[15]

A coupled neutronics-thermal-hydraulics analysis for the HPLWR core has been developed to identify hot spots in the fuel assemblies and to verify that the material limits can be met. With these coupled codes, axial and radial power distribution of the SCWR core will be determined as well as temperature distributions for cladding, moderator and coolant.

III.B. Safety concept for the HPLWR

A safety concept for the HPLWR has been proposed by Bittermann *et al*,[16] which now needs to be worked out by means of analyses of transient and accident scenarios using state-of-the art system codes. Plant models including an accurate representation of the reactor pressure vessel (RPV) and the complex geometry of the core (three-pass core) have been developed for various thermal-hydraulic codes (RELAP5, CATHARE, APROS) as well as for coupled neutronic thermal-hydraulic codes (SMABRE/TRAB-3D, ATHLET-KIKO3D). The initial analyses focused on the hydraulics of the initial core design, which contributed to the development of an improved geometry. The analyses of the current configuration showed that the flow can now be expected to be adequate for all loads of interest. Results concerning these aspects will be published in due course.

A list of transient and accident scenarios has been compiled, which includes the supposedly most severe conditions with respect to preserving the fuel integrity. As a continuous coolant mass flow rate through the reactor is required for the once-through steam cycle implemented in the design of the HPLWR, special attention must be devoted to transients resulting in a loss of flow. Analyses are currently carried out to verify the response of the plant to a Loss of Feed Water (LOFW), including various cases with respect to pump failures, run-down times and scram intervention times. These studies aim, among others, to determine the required features of the feed-water pump-motor system, including the need for fly-wheels. Loss-Of-Coolant Scenarios have been started to be analyzed, and preliminary results have recently been obtained. These analyses are especially important because of the cool ability of the three-pass core following a sudden depressurization and the consequent fast reduction in water inventory in the core is a critical issue for the safety concept, as no natural circulation mechanism is available.

A preliminary safety system design has been proposed by de Marsac *et al*,[17] which has to be specified now in more detail to control the individual accident scenarios. For instance, the low pressure coolant injection (LPCI) system could be similar to the active LPCI system of a boiling water reactor. It could act either after depressurization through the spargers or in case of a loss of coolant accident. Investigations are currently under way for the optimization of the active heat removal systems. Additionally, a passive high pressure coolant injection (HPCI) system is investigated, which has not yet been applied to any pressurized or boiling water reactors. Detailed transient analyses will be required to decide about its feasibility.

Keresztúri *et al*,[18] started to simulate RIAs scenarios, and the results obtained show significant perturbation of the local power. Due to the very heterogeneous moderator density distribution, the results are sensitive to a great extent to the initial position of the control rods and to core loading, especially to the number and position of the assemblies containing burnable poison rods. The acceptance criteria, however, are fulfilled so far. In some cases, the hot channel temperatures are not far from the limits, which points out the necessity of the RIA analyses. In the next period, a broad range of RIAs and ATWS transient will be investigated, and the analyses will show whether modifications of the core neutronic design will be required.

IV. STUDY OF HEAT TRANSFER TO SUPERCRITICAL PRESSURE FLUID IN JAPAN

A conceptual study of the pressure-vessel type SCWR started at the University of Tokyo in 1989. The GIF SCWR concept with pressure-vessel is based on the concept that has been developed in the University of Tokyo.[19]

Two R&D projects on the pressure-vessel type SCWR with fast/thermal options are ongoing in Japan jointly by universities, research institutes and industries.[20]

A R&D project on fast option, entitled "Research and development of the Super Fast Reactor" was entrusted to the University of Tokyo in December 2005 and will be completed in March 2010. Aiming at a highly economical fast reactor, the plant concept is being developed with quantitative characteristics/performances. The databases of the thermal hydraulics and materials (including water chemistry) are being developed by experiments.

Another R&D project on thermal option, entitled "Development of SCWR in GIF Collaboration (Phase-I)", was granted to Toshiba Corporation and The Institute of Applied Energy in August 2008 and will be completed in March 2011. The purpose is to assess the viability of the thermal SCWR concept through the GIF collaboration. The R&D areas include System Integration and Assessment, Thermal-Hydraulics and Safety, and Materials and Chemistry, which correspond to the GIF/SCWR projects.

In this report the typical thermal-hydraulic test results for fast option are explained.

IV.A. Heat Transfer test for Freon[21]

Thermal-hydraulics tests at supercritical pressure conditions with water and Freon have been done to obtain heat transfer data, using tube and bundle.

Experiments are performed with a supercritical pressure HCFC22 forced circulation loop, newly set up at Kyushu University, Japan. HCFC22 is used as a substitute for water because its critical pressure and temperature of 4.99 MPa and 96.2°C are far lower than those of water (22 MPa and 374°C), and therefore the experimental conditions can be flexibly altered. Steady state tests are carried out with a single circular tube test section of 4.4.mm I.D and with a sub-bundle (Bundle-I) test section composed of seven heater rods simulating the actual fuel bundle geometry as shown in Figure 3.

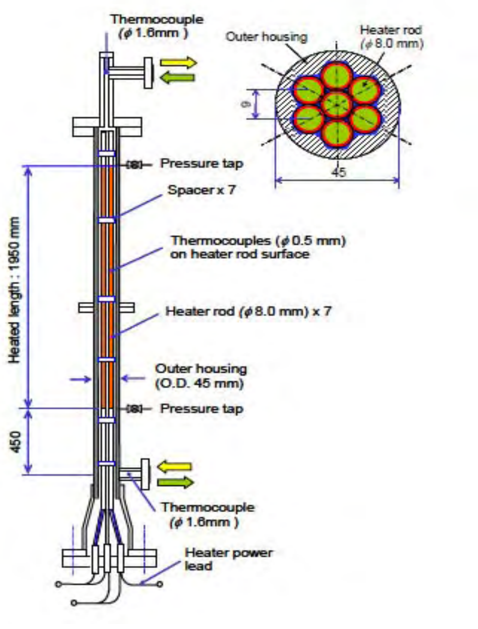

Figure 3: Sub-bundle(Bundle) test section

Figure 4 shows the typical result. In the sub-bundle channel, the occurrence of heat transfer deterioration is generally suppressed even in the upward flow, and the heat transfer is similar to that in the tube flow in the normal heat transfer region of the tube flow.

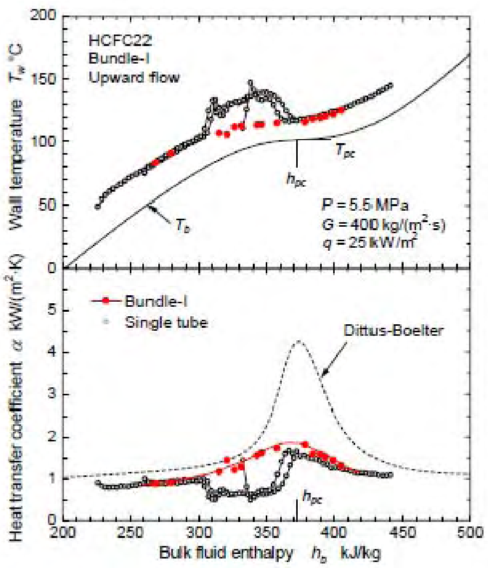

Figure 4: Typical comparison between tube and Bundle

IV.B. Heat Transfer test for supercritical pressure water[22]

Figure 5 shows a schematic drawing of the test section. A test section including a single heater rod simulating fuel rod has been fabricated and installed into a high pressure and high temperature water circulation loop in JAEA. Supercritical pressure water at 25 MPa flows in the test section and the surface temperatures of the heater rod are measured to evaluate the heat transfer coefficient around the single heater rod.

Figure 6 shows the typical data for wall temperatures and heat transfer coefficients. HTC in this Figure is the heat transfer coefficient evaluated by Dittus-Boelter correlation based on the hydraulic diameter of the test section. Run930 and Run698 are the experimental results obtained by Yamagata *et al.*,[23] using a straight tube with the ID 10 mm at the mass flux of 1156-1235 kg/m²/s and the heat flux of 698 and 930 kW/m², respectively. The measured maxi–mum heat transfer coefficient in this Figure is about 29 kW/m²/K. The maximum value and the bulk fluid enthalpy for which the maximum value occurred are lower than the predicted values. This trend is almost the same as the Yamagata's Run 930.

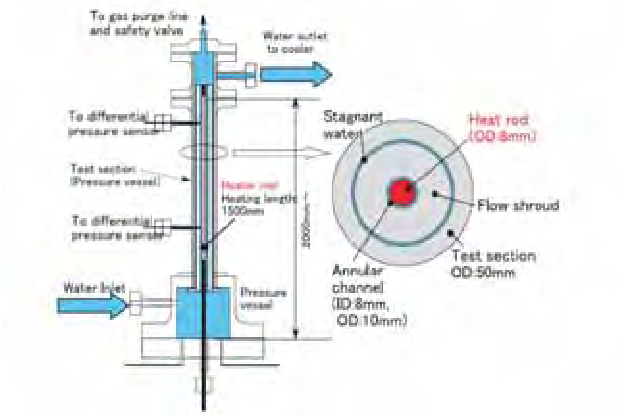

Figure 5: Schematics of test section for supercritical water around a single heater rod

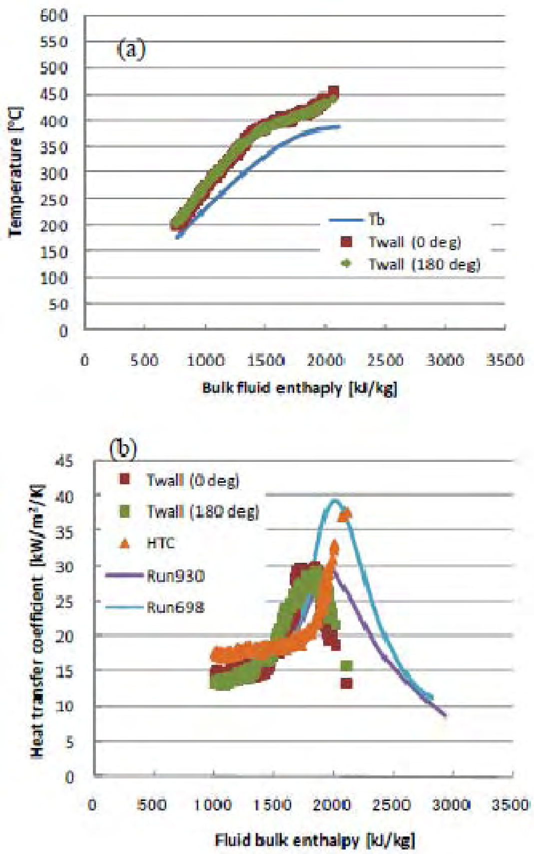

Figure 6: Typical test results for power=40 kW, G=2060-2480 kg/m²/s P=25.5 MPa

V. KOREAN CONTRIBUTION TO THE GIF THERMAL-HYDRAULICS AND SAFETY PROJECT

V.A. Heat Transfer test for CO_2

Figure 7 shows a schematic diagram of the test facility. The design pressure and temperature of the main loop are 12.0 MPa and 80°C, respectively. Figure 8 shows the test sections and the locations of the measuring points. The test section at the left is a circular tube with an inside diameter of 4.4 mm and heated by a direct current power supply to impose a uniform heat flux on the tube internal surface. The middle one is a 9 mm tube test section.

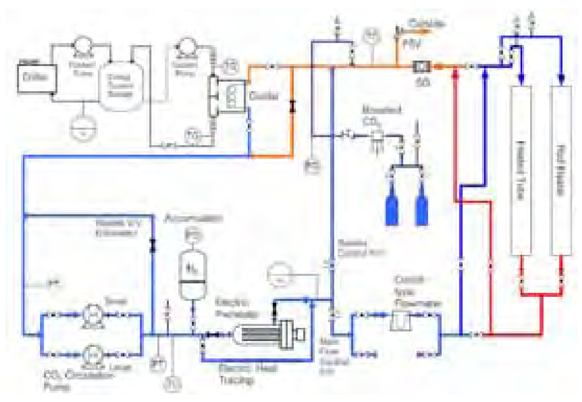

Figure 7: Schematic of test facility SPHINX

The details are the same as those for the 4.4 mm tube and 6.32 mm tube, except for the heated length. The tube of 6.32 mm ID corresponds to the subchannel hydraulic diameter of the core design by KAERI.[24] The right one is the test section for an annular channel. A heater rod with an outside diameter (OD) of 8 mm is centered in the 10 mm ID tube. The hydraulic diameter of this annulus channel is the same as the 4.4 mm tube.

The supercritical CO_2 flows upward. The fluid temperatures are measured in the mixing chambers at the inlet and the outlet of the test section as well as along the tube surface. An eccentric annular subchannel of 9.5 x 12.5 mm (1 and 2 mm gaps for the narrow and wide side, respectively) was tested also.[25]

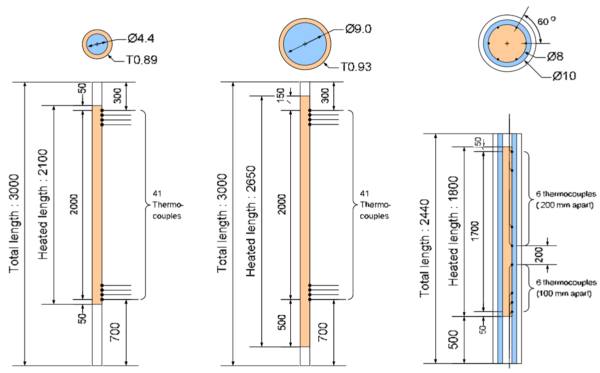

Figure 8: Test sections (tube and annulus channel)

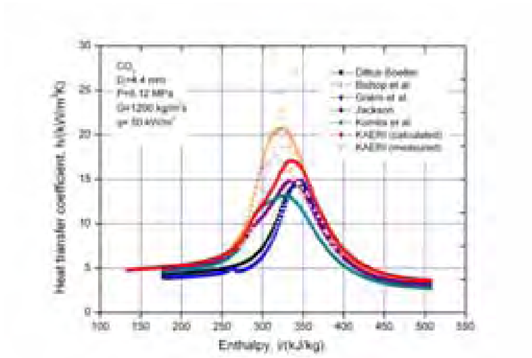

Figure 9: Comparison of the estimated heat transfer coefficient by various correlations against the experimental data.

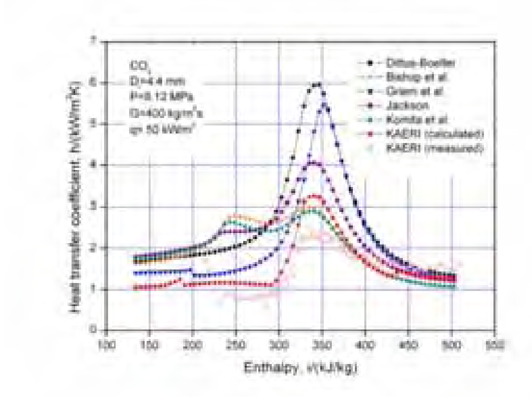

Figure 10: Comparison of the estimated heat transfer coefficient by various correlations against the experimental data.

Figure 9 and Figure **10** show typical results of the research on a heat transfer at supercritical pressure at KAERI. In the Figures the predictions by various correlations including the correlation, which was proposed based on the test data obtained at KAERI, are shown. At a condition of p=8.12 MPa, q=50 kW/m^2 in the tubes of 4.4, mm ID the proposed correlation predicts the experimental data reasonably well when compared with the existing correlations for both the normal and impaired conditions.

Table **1** summarizes the geometries and dimensions of the test sections used in the experiments at SPHINX in KAERI. The test results have been published elsewhere.[24-31]

In 2009, downward flow tests for the annulus channels will be performed.

Geometry and dimension			Up flow	Down flow
Tube	4.4 mm		●	●
	6.32 mm	Plain	●	●
		Wire	●	●
	9.0 mm		●	●
Concentric annulus (8 x 10 mm)			●	2009
Eccentric annulus (9.5 x 12.5 mm)			●	2009
Pressure transient			2009	2009

Table 1: Tested or planned geometries and dimensions at SPHINX in KAERI

V. B. Adaptation of Safety Analysis Code

Based on TASS/SMR[32], a computer code, TASS/SCWR has been developed for the safety analysis of a SCWR. For the modelling of a reactor coolant system, five one-dimensional conservation equations of a two-phase flow are formulated,[33] where the thermodynamic properties are calculated by using the IAPWS-IF97[34] formulation. The fission power input to the fuel is obtained from the reactor kinetics equations with six delayed neutron groups. An ANS73 decay heat curve has been incorporated into the database. Heat transfer correlations and conduction equations are modelled for the calculation of a heat generation in a core and the heat removal in a passive residual heat removal system (PRHRS).

A natural circulation condition has been simulated by using the TASS/SCWR and MARS[35] codes, and the results were compared with each other. Initial pressure and temperature were 25 MPa and 300°C. 10 MW was supplied to node 2 and removed from node 12. The fluid density and temperature calculated by TASS/SCWR and MARS, as shown in Figure 11 were in good agreement with each other.

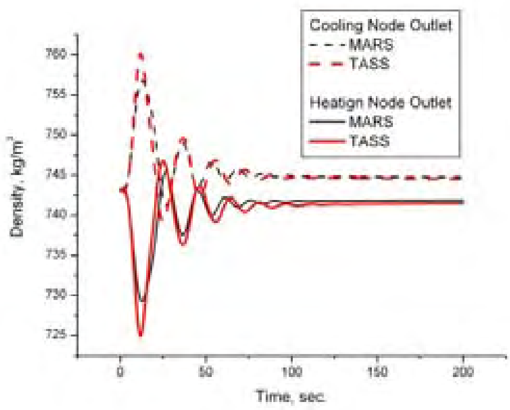

Figure 11: Comparison of the density between TASS/SCWR and MARS.

IV. CONCLUSION

In this paper the ongoing research activities as well as recent accomplishment in the TH&S PMB member countries are provided. Although the official endorsement of PA is still under processing, relatively active research is being performed in the member countries as well as in China (not introduced here since its participation status is currently an observer). The degree of research depth in the member countries is different from each other, and it will be relieved by continuing negotiation and further collaboration process.

A number of research and development (R&D) projects have been established to provide thermal-hydraulics and safety-related informa–tion in support of the development of the CANDU® supercritical water-cooled reactor (SCWR) in Canada. Thermal-hydraulics-related projects in these programs focus on the development of heat transfer prediction methods for CANDU-type bundles. These projects cover analytical and experimental studies in supercritical water and surrogate fluids in tubes and bundles.

Two R&D projects on the pressure-vessel type SCWR with fast/thermal options are ongoing in Japan jointly by universities, research industries and industries.

In Europe, the High Performance Light Water Reactor (HPLWR) is currently under development. The High Performance Light Water Reactor is a Light Water Reactor (LWR) with supercritical water at 25 MPa as coolant and moderator. Computational fluid dynamics, analytical work, and coupled neutronic-thermal hydraulic analyses have been executed to assess the heat transfer rate and fuel temperature in the core of the HPLWR. A safety concept for the HPLWR has been proposed, which now needs to be worked out by means of the analyses of transient and accident scenarios. For this purpose several thermal-hydraulic system codes have been upgraded and tested for super critical water conditions.

Korea's 3-year project comes closer to the end as of February 2010. The work scope has been limited to the supercritical heat transfer to CO_2 due to the lack of resources. The continuing support of the government for the fundamental research on SCWR is expected at least in the fields of thermal-hydraulics and safety, and material and chemistry.

Acknowledgements

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Japans' contribution includes the results of "Research and Development of the Super Fast Reactor" entrusted to the University of Tokyo by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT), and the results of "Development of SCWR in GIF Collaboration (Phase-I)," funded by the Ministry of Economy, Trade and Industry (METI).

Korea's contributions are part of the results of the 3-year project (2008-2010) in the Nuclear R&D program funded by the Ministry of Education, Science and Technology.

Nomenclature

d	diameter
T	temperature
P	pressure, MPa
G	mass flux, kg/m^2s
q	heat flux, kW/m^2

Subscript

b	bulk
w	wall

References

1. Brady, D., R.B. Duffey, H. Khartabil, R. Sadhankar, and S. Suppiah, "Generation IV Reactor Development In Canada", 3rd Int. Symposium on SCWR, Shanghai, China, March 12-15 (2007).

2. Pioro, I.L. and R.B. Duffey, "Experimental Heat Transfer to Supercritical Water Flowing inside Channels (Survey)," Nucl. Eng. Des. Vol. 235, No. 22, pp. 2407-2420, (2005).

3. Leung, L.K.H., D.C. Groeneveld, P.R.S.S Gudla., and S. Tavoularis, "An Expanded Database on Supercritical Heat Transfer for Water, Carbon Dioxide and Other Fluids", *Proc. 7th International Topical Meeting on Nuclear Reactor Thermal-hydraulics, Operation and Safety (NUTHOS-7),* Seoul, Korea, October 5-9, (2008).

4. Groeneveld, D.C, S. Tavoularis, P.R.S.S. Gudla, S.K. Yang, and L.K.H. Leung, "Analytical and Experimental Program of Supercritical Heat Transfer Research at the University of Ottawa," Nucl. Eng. Techn., Vol. 40, No. 2, pp. 107-116, (2008).

5. Chatoorgoon, V., M. Shah, R. Duffey, "Linear Predictions of Supercritical Flow Stability in Parallel Channels", *Proc. 3rd Int. Symposium on SCWR* – Design and Technology, Shanghai, China, March 12-15, (2007).

6. Starflinger, J., T. Schulenberg, P. Marsault, D. Bitterman, C. Maraczy, E. Laurien, J.A. Lycklama à Nijeholt, H. Anglart, N. Aksan, M. Ruzickova and L. Heikinheimo, "European Research Activities within the Project: "High performance Light Water reactor Phase 2" (HPLWR phase 2)", *Proc. of ICAPP'07*, Nice, France May 13-18, paper 7146, (2007).

7. Loewenberg, M.F., *et al.*, "Supercritical Water Heat Transfer in Vertical Tubes: a Look-up Table", *Progress in Nuclear Energy*, Vol. 50, 532-538, (2008).

8. Zhu, Y., and E. Laurien, "Predictions of Heat Transfer of Upward Flow in Annular Channel at Supercritical Pressure – water and CO_2", *Proc. of the 4th International Symposium of Supercritical Water Cooled Reactors*, Heidelberg, Germany (2009).

9. Visser, D.C., J.A. Lycklama à Nijeholt and F. Roelofs, 2008a, CFD Predictions of Heat Transfer in the Super Critical Flow Regime, *ICAPP 2008,* Anaheim USA, Paper 8155, (2008).

10. Palko, D., and H. Anglart, "Investigation of the Onset of Heat Transfer Deterioration to Supercritical Water", *Proc. of the 4th International Symposium of Supercritical Water Cooled Reactors*, Heidelberg, Germany (2009).

11. Anglart, H., "Heat Transfer Deterioration in Application to HPLWR-Mechanisms Identification and Ranking Table", *Proc. of the 4th International Symposium of Supercritical Water Cooled Reactors,* Heidelberg, Germany (2009).

12. Himmel S., *et al*., "Determination of Mixing Coefficients in a Wire-Wrapped HPLWR Fuel Assembly using CFD", *Proc. of ICAPP 2008,* Anaheim USA, Paper 8053, (2008).

13. Chandra, L., J.A. Lycklama à Nijeholt, D.C. Visser and F. Roelofs, "CFD analyses of the influence of wrap spacers on the heat transfer at super critical conditions", *Proc. of the 4th International Symposium of Supercritical Water Cooled Reactors*, Heidelberg, Germany (2009).

14. Laurien, E., *et al*., "Flow and Heat Transfer of a Heated Rod with a Wrapped Wire inside a Channel", *Proc. of the 4th International Symposium of Supercritical Water Cooled Reactors,* Heidelberg, Germany (2009).

15. Kiss, A., E. Laurien and A. Aszodi "Numerical Simulation of a HPLWR Fuel Assembly Flow with Wrapped Wire Spacers", *ICAPP Conf. 2008,* Anaheim, CA USA, June 8-12, Paper No. 8339, (2008).

16. Bittermann, D., T. Schulenberg, M. Andreani, "The Safety Concept of the HPLWR", 4th International Symposium on Supercritical Water-Cooled Reactors, March 8-11, Heidelberg, Germany, (2009).

17. De Marsac, B., D. Bittermann, J. Starflinger, T. Schulenberg, "Containment Design Proposal with Active and Passive Safety Systems for a High Performance Light Water Reactor", 4th International Symposium on Supercritical Water-Cooled Reactors, March 8-11, Heidelberg, Germany, (2009).

18. Keresztúrun, A., Cs. Maráczy, I. Trosztel, Gy. Hegyi, "Safety Analyses of Reactivity Initiated Accidents in a HPLWR reactor by the Coupled ATHLET-KIKO3D code", 4th International Symposium on Supercritical Water-Cooled Reactors, March 8-11, Heidelberg, Germany, (2009).
19. Oka, Y., *et al*., Research Program on Super Fast Reactor, Proc. ICAPP'06, Reno, NV, USA, June 4-8, 2006, Paper 6353 (2006).
20. Ishiwatari, Y., *et al*., JAPANESE R&D PROJECTS ON PRESSURE-VESSEL TYPE SCWR, 4th Int. SCWR Conf., Paper No. 73 (2009-3).
21. Mori, H., *et al*., Research and Development of a Super Fast Reactor (7), proc. 16PBNC, Aomori, Japan, Oct. 13-18, (2008).
22. Ezato, K., *et al*., Research and Development of a Super Fast Reactor (8), proc. 16PBNC, Aomori, Japan, Oct. 13-18, (2008).
23. Yamagata, K., K. Nishikawa, S. Hasegawa, T. Fujii, and S. Yoshida, "Forced Convective Heat Transfer to Supercritical Water Flowing in Tubes," Int. J. Heat and Mass Transfer, Vol. 15, pp. 2575-2593, (1972).
24. Kim, H.Y., Y.Y. Bae, "Experimental Study on Heat Transfer to Supercritical CO2 in a Circular Tube," 16th Pacific Basin Nuclear Conference (16PBNC), Aomori, Japan, Oct. 13-18, (2008).
25. Bae, Y.Y., H.Y. Kim, "Heat transfer at supercritical pressure in an eccentric annulus channel," NTHAS6: Sixth Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety Okinawa, Japan, November 24- 27, (2008).
26. Kim, H.Y., H. Kim, J.H. Song, B.H. Cho, Y.Y. Bae, "Heat transfer test in a vertical tube using CO_2 at supercritical pressures," *Journal of Nuclear Science and Technology*, 44 (3) 1–9 (2007).
27. Kim, H.Y., H. Kim, D.J. Kang, J.H. Song, Y.Y. Bae, "Experimental investigations on heat transfer to CO2 flowing upward in a narrow annulus at supercritical pressures," *Nuclear Engineering and Technology*, 40 (2) 155–162. Special Issue on the 3rd International Symposium on SCWR (2008a).
28. Kim, H., H.Y. Kim, J.H. Song, Y.Y. Bae, "Heat transfer to supercritical pressure carbon dioxide flowing upward through tubes and a narrow annulus passage," *Progress in Nuclear Energy*, 50 518–525 (2008b).
29. Kim, H., Y.Y. Bae, H.Y. Kim, J.H. Song, B.H. Cho, "Experimental investigations on heat transfer characteristics in upward flow of supercritical carbon dioxide," *Nuclear Technology*, 164 119–129 (2008c).
30. Bae, Y.Y., H.Y. Kim, "Convective heat transfer to CO_2 at a supercritical pressure flowing vertically upward in tubes and an annular channel," *Experimental Thermal and Fluid Science* Vol. 33, Issue 2, 329-339 (2009a).
31. Bae, Y.Y., H.Y. Kim, T. Yoo, "Heat transfer experiments in a wire-inserted tube at supercritical conditions," *4th International Symposium on Supercritical Water-Cooled Reactors March 8-11, 2009, Heidelberg, Germany* (2009b).
32. Yoon, H.Y., *et al*., Thermal hydraulic model description of TASS/SMR, KAERI/TR-1835/2001, Korea Atomic Energy Research Institute, (2001).
33. Yoon, H.Y., S.H. Kim, H.C. Kim, Y.Y. Bae, "Development of a Safety Analysis Code for a Supercritical Water Cooled Reactor (SCWR)," *Transactions of the Korean Nuclear Society Autumn Meeting* Busan, Korea, October 27-28, (2005).

34. Parry, W.T., J.C. Bellows, J.S. Gallagher, and A.H. Harvey, "ASME International Steam Tables for industrial use: Based on the IAPWS International Formulation 1997 for the thermodynamic Properties of Water and Steam (IAPWS-IF97)," CRTD- 58, ASME, (2000).

35. Chung, B.D., *et al*., MARS3.0 Code manual input requirements, KAERI/TR-2811/2004, Korea Atomic Energy Research Institute, (2004).

SCWR MATERIALS AND CHEMISTRY
STATUS OF ONGOING RESEARCH

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I. INTRODUCTION

The idea of using a supercritical water (SCW) coolant in a water-cooled reactor dates back to the 1960s, [1, 2] although no such reactor was ever built. More recently, two types of supercritical water-cooled reactor (SCWR) concept have evolved from existing light water reactor (LWR) and pressurized heavy water reactor (PHWR) designs: (a) a number of designs [3, 4, 5] consisting of a large reactor pressure vessel containing the reactor core (fueled) heat source, analogous to conventional Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) designs, and (b) designs with distributed pressure tubes or channels containing fuel bundles, analogous to conventional CANDU[1] and Reactor Bolshoy Moschchnosty Kanalny (RBMK) nuclear reactors. [6] Designs in both concepts are typically direct cycle, with out-of-core portions similar to existing fossil-fired generators. Aside from the design concept itself, the most important technical issue is likely to be the identification of a) materials for in-core and out-of-core components and b) an appropriate coolant chemistry. The reference design for the SCWR [7, 8] calls for an operating pressure of 25 MPa, a core inlet temperature of about 280°C and a core outlet temperature up to 620°C. Peak fuel cladding temperatures could be as high as 850°C in some designs. [9]

The SCW coolant in both the pressure vessel and pressure tube concepts lies in both the liquid and supercritical fluid areas of the T-P phase diagram (Figure 1). The coolant will pass through the critical point at some location in the reactor core. The corrosivity of SCW varies widely depending upon the values of properties such as the density, ion product and dielectric constant, as well as on the nature of any solutes present (impurities, dissolved oxygen) and their concentrations. [10] At the low density (~0.1 g/cm^3) expected at the core outlet of an SCWR, SCW is a non-polar solvent able to dissolve gases like oxygen to complete miscibility. While the solubility of ionic species is expected to be extremely low under these conditions, the formation of neutral complexes increases with temperature, and can become important under near-critical and super-critical conditions. It has been suggested that the most important temperature range is from 275 to 450ºC, over which the properties of water change dramatically, and solvent compressibility effects exert a huge influence on solvation. With the exception of a few recent studies, [11] the thermochemistry of neutral hydrolysed metal species is poorly understood, even at temperatures well below the critical point.

[1]CANDU® **CAN**ada **D**euterium **U**ranium, is a registered trademark of Atomic Energy of Canada Limited (AECL).

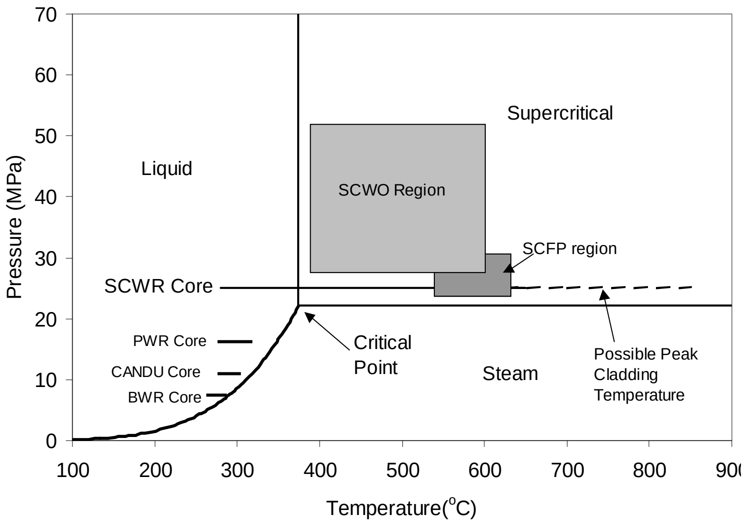

Figure 1: Temperature-pressure phase diagram of water. The operating regions of present BWR, PWR and CANDU plants and of proposed SCWR designs are indicated. Also shown are the operating regions for supercritical fossil-fired boilers (SCFP) and supercritical water oxidation (SCWO) processes. Adapted from [12].

The GIF SCWR materials and chemistry provisional project management board (PPMB) has identified two major challenges that must be overcome to ensure the safe and reliable performance of an SCWR:

1. Insufficient data are available for any single alloy to unequivocally ensure its performance in an SCWR, especially for alloys to be used for in-core components.

2. Current understanding of SCW chemistry is inadequate to specify a chemistry control strategy, as the result of the large changes in physical and chemical properties of water through the critical point, coupled with the as yet poorly understood effects of water radiolysis.

To address these challenges, two Work Packages, one on SCWR Materials and the other on Radiolysis and Water Chemistry, have been developed for the GIF SCWR Materials and Chemistry draft Project Plan. This paper broadly outlines these work packages, describes some of the key challenges, and presents some of the progress being made to overcome these challenges.

II. MATERIALS

Although they have different requirements for most core components (*e.g.*, reactor pressure vessel internals, nozzles, supports in a pressure vessel design; ceramic insulator, metallic liner in a pressure tube design), the pressure vessel and pressure tube designs share common issues with respect to materials for out-of-core components and fuel cladding. There are therefore strong synergies between the materials R&D needs of the two designs.

Initial alloy selection for testing for the SCWR was guided by existing data from supercritical and ultra-supercritical fossil-fired power plants and supercritical water oxidation (SCWO) systems. While extensive testing was carried out in support of the development of SCWO processes, the chemistry conditions were typically not of direct relevance to an SCWR, being very acidic with high concentrations of aggressive species such as chloride ion. Therefore, although knowledge gained from current reactor designs, modern boiler technologies and research in support of SCWO has provided valuable insights that have aided in the identification of key parameters, the PPMB has concluded that there are still significant gaps in our knowledge about the properties of the materials under proposed SCWR operating conditions.

To address these gaps the draft Project Plan divides the work package on materials into two main tasks:

1. Study of Un-irradiated Materials
2. Study of Irradiated Materials

Each main task has been further sub-divided into the following sub-tasks:

1. Development of Materials Data-bases in SCW
2. Material Testing and Performance Evaluation
3. R&D on Coatings and Surface Modification

A four tier testing strategy has been developed (Figure 2), reflecting the fact that fundamental data on relevant materials properties can be measured with great precision and a high degree of control in test loops and autoclaves (Figure 2, Levels 1-3). Tests using irradiated materials or simulated radiolysis conditions significantly increase the experimental challenges. Complete control of all test parameters will be difficult in an in-reactor test loop (Figure 2, Level 4). For example, direct measurement of the electrochemical corrosion potential (ECP) of a test specimen in an in-core loop is not possible with existing technologies.

Sub-task 2 in the Project Plan (Material Testing and Performance Evaluation) focuses on acquiring data on, and developing a mechanistic understanding of, the following key material properties:

1. Corrosion and Stress Corrosion Cracking (SCC)
2. Dimensional and microstructural stability
3. Strength, embrittlement and creep resistance.

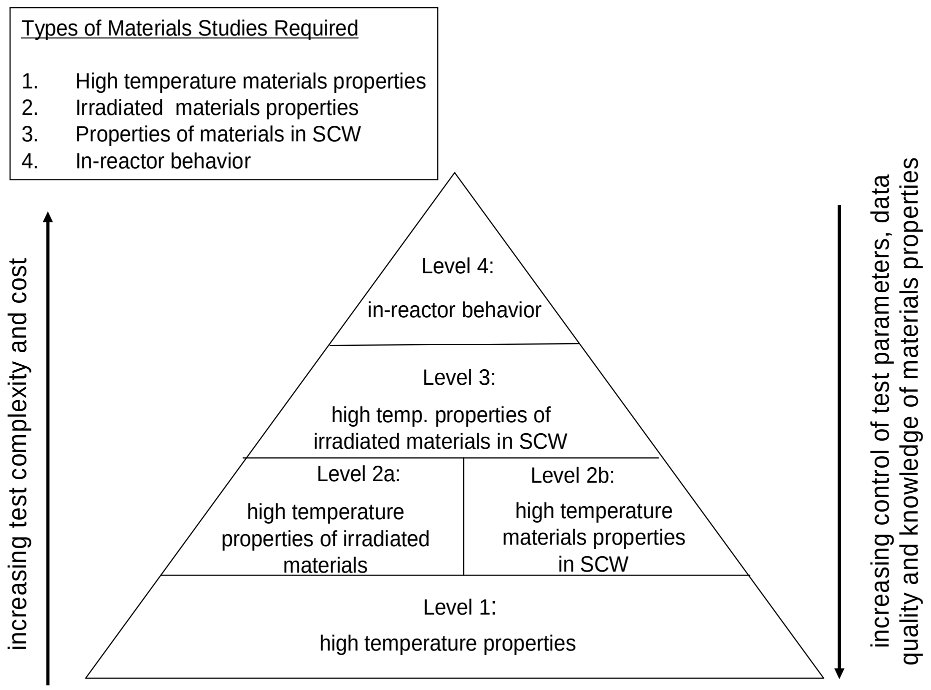

Figure 2: Schematic of the four-tier test program for the GIF SCWR Materials Work Package.

Alloy class	Temperature (°C)	Water chemistry	Exposure time (h)
Austenitic SS	290-650	DO[a] from deaerated[b] to 8 ppm	24-3000
Ni-base	290-600	DO from deaerated to 8 ppm, Conductity <0.1 mS/cm	24-3000
Ferritic-Martenstic	290-650	DO from deaerated to 8 ppm, Conductity <0.1 mS/cm	100-3000
Oxide Dispersion Strengthened steels	360-600	25 ppb	200–3000
Zr-base	400-500	Deaerated DO, Conductity <0.1 mS/cm	<2880
Ti-base	290– 550	8 ppm DO, Conductity 0.1 mS/cm	500

a – dissolved oxygen
b – typically <10 ppb

TABLE I: Summary of Materials Corrosion Testing Under SCWR Conditions (adapted from [14])

Table I summarizes the alloy classes and test conditions examined to date. The alloy classes tested include ferritic-martensitic (F/M) and austenitic steels, Ni-base alloys, Oxide Dispersion Strengthened (ODS) steels, Zr-base alloys, and Ti-base alloys; most of the focus has been on the first four classes. Several alloys have been the subject of numerous studies (*e.g.*, Alloy 625), but some data under relevant conditions are currently available for approximately 90 alloys. The available corrosion data under SCWR conditions show that the oxidation rate of steels, especially F/M steels, is rather high, increasing rapidly above 500°C. In addition to general corrosion, SCC (intergranular and transgranular) is expected to be a critical degradation mode in an SCWR [13, 14]. The mechanism of SCC in SCW is currently being studied using techniques such as slow strain rate testing and U-bend specimens; more sophisticated loading methods are also being developed (*e.g.*, [13]). SCC requires both a mechanical and a chemical component (*e.g.*, T, P, water chemistry, loading mode, material processing), so that a large number of parameters must be studied. The existing data show that austenitic stainless steels and Ni-base alloys exhibit greater susceptibility for SCC than F/M alloys.

The key experimental variables affecting corrosion identified to date are temperature, water density (pressure), dissolved oxygen concentration, water conductivity and surface finish. Tests have been performed at temperatures ranging from below the critical temperature up to 650°C. The water chemistry has typically been low conductivity “pure” water with dissolved oxygen concentrations ranging from <10 ppb to 8 ppm. Test durations have ranged from 24 to 3 000 hours. After exposure to SCW, test specimens have been characterized using techniques ranging from weight change to surface analytical methods such as Scanning Electron Microscopy, Transmission Electron Microscopy and Scanning Auger Microscopy. Tests have been performed in static autoclaves, capsules, and loops; each type of test facility has advantages and disadvantages. To facilitate comparison of data from different laboratories and tests facilities, a series of round robin tests are planned, commencing in 2009, using a standard set of test conditions (Table II) and coupon preparation procedures.

To assist in interpretation of the large amounts of data now becoming available for some alloys, a key project task (Sub-task 1) is the development of a database of corrosion and materials data. Data for the database will be taken from the existing literature in addition to data from experiments underway as part of the project.

Parameter	Value
Temperature (°C)	550
Pressure (MPa)	25
Dissolved Oxygen Concentration (ppb)	50 and 8000
$pH_{25°C}$	~7 at room temperature and pressure
Water resistivity, room temperature and pressure (MΩ)	18
Test duration (h)	500
Test Alloys	310 SS, Alloy 690, P/T 91, ODS (MA956, MA957)

TABLE II: Test Conditions for Round Robin Testing

Work is underway to characterize the mechanical properties of candidate alloys, including fracture toughness, tensile strength, and creep resistance. For in-core materials, understanding irradiation-induced changes due to growth, swelling, helium-bubble formation, dislocation microstructure, precipitate micro-structure and irradiation-induced composition changes, and demonstrating that these changes will not compromise the integrity of core components is a key requirement. Some measurements on irradiated materials have been carried out, [15] and much more work is needed in this area.

In addition to experimental programs, focused modelling effort is underway to improve understanding of materials-environment interactions within a shorter time frame. A number of key degradation processes (*e.g.*, general corrosion, pitting, SCC initiation and growth, irradiation and thermal creep) are being modelled using the latest computational techniques.

As no one alloy has yet been identified that possesses all of the properties required for critical SCWR applications (good resistance to corrosion at the surface, good resistance to SCC, creep and radiation damage in the bulk), one potential solution is to modify the surface of a material possessing the required bulk properties to impart the desired corrosion resistance [16, 17]. This approach is being explored in the third materials sub-task. While only limited work has been carried out on surface modification, there have been some promising results; for example, because the surface alloying layer can be produced after fabrication of in-core components (*i.e.*, after forming and joining operations), formability and weld ability issues can be avoided.

III. CHEMISTRY

There is a strong interplay between coolant chemistry and materials selection in any water-cooled nuclear power plant system. The long-term viability of an SCWR will depend on the ability of reactor developers to identify a set of chemistry control specifications that will satisfy the (sometimes conflicting) requirements to minimize materials degradation and radionuclide transport, optimize thermal performance and maximize system lifetime. The SCWR coolant will undergo a transition from “water-like” to “steam-like” densities (from ~0.8 to 0.1 g/cm^3) as it passes from subcritical to supercritical conditions through the reactor core. Operating experience from supercritical thermal power stations has shown that the region of most importance is the “pseudo transition zone” [18] from 275 to 450°C at 25 MPa. Few quantitative studies of aqueous solutes have been performed above 300°C; above 450°C, SCW is sufficiently steam-like that solid-gas thermodynamic models may be adequate.

Compared to the large body of work on materials testing, little work on SCWR water chemistry has yet been carried out. [19, 20, 21] The long-term goal of the Radiolysis and Water Chemistry Task is to specify a suitable water chemistry for the SCWR design. Candidate water chemistry regimes and specifications for key chemistry parameters (pH, dissolved oxygen and hydrogen concentrations, concentrations of any other additives, allowable concentrations of impurities, etc.) must be identified prior to any long-term materials testing.

The Radiolysis and Water Chemistry work package consists of four Tasks:

1. Studies of Radiolysis of SCW.
2. Understanding Corrosion Product Transport and Deposition.
3. Specification of Water Chemistry for Detailed Testing.
4. Identification of Methods for Chemistry Monitoring and Control.

As experiments at very high temperatures and pressures, especially beyond the critical point of water, are difficult to perform, computer simulations are an important route of investigation for Tasks 1 and 2. However, a large amount of fundamental experimental data will be needed in order to develop such models, and the model predictions will need to be further validated against experimental data.

Potentially the biggest challenge for the development of an SCWR water chemistry regime will be predicting and mitigating the effects of water radiolysis. [22-24] The radiolytic production of oxidizing species (*e.g.*, OH, H_2O_2, O_2, and HO_2/O_2^-) can increase corrosion of reactor components as well as affecting corrosion product transport and deposition. While current PWRs and PHWRs limit the formation of oxidizing species by ensuring the presence of excess hydrogen at concentrations sufficient to chemically minimize the net production of oxidizing species by radiolysis, there are insufficient data to determine whether this strategy would be effective in an SCWR. As a consequence the coolant could be very oxidizing immediately downstream of the core. Work is on-going to develop an improved understanding of SCW radiolysis through a combination of experiments and modeling.

The release and transport of corrosion products (CPs) from the surfaces of system components has been a serious concern for all water-cooled nuclear power plants. High levels of CP transport can result in: a) increased deposition on fuel cladding surfaces, leading to reduced heat transfer and the possibility of fuel failures, and b) increased production of radioactive species by neutron activation, ultimately increasing out-of-core radiation fields and worker dose. In addition, nuclear and thermal power stations experience deposition of steam-volatile species on turbines at levels that can cause turbine failure. Supercritical thermal station experience suggests CP deposition could be significant in an SCWR. [25, 26] Some preliminary work on CP transport in an SCWR has been performed [20, 26] with encouraging results. [27]

Several water chemistry regimes are typically used in fossil-fired SCW plants [26, 28] (Table III). However, to date most experimental work on SCWR materials has been carried out in a limited range of water chemistries, namely pure water, pure water with added oxygen (50-8 000 ppb), and hydrogen water chemistry (H_2 concentration ~ 30 cm^3/kg water). Additional testing under a wider range of water chemistries is required. These candidate water chemistries will need to be assessed in an in-reactor loop, such as the loop currently being commissioned at the Nuclear Research Institute Řež plc in the Czech Republic, as a part of the European Union High Performance Light Water Reactor project, [29] to determine their effect on radiolysis and corrosion product transport.

It will be necessary to monitor and control relevant chemistry parameters (*e.g.*, conductivity, pH, ECP, concentrations of dissolved hydrogen and oxygen) in an SCWR and in in-reactor test loops. [19] Existing methods of chemistry monitoring, predominantly ex-situ (cooled and de-pressurized) and off-line (laboratory analysis of grab samples), will be inadequate in an SCWR, as a result of the large changes in water chemistry around the critical point. It is likely that reliable monitoring of key chemical parameters can only be achieved through the development of in-situ or on-line probes, and there is a need for more work on this topic.

Table III: Summary of Water Treatments used in Supercritical Water Fossil-Fired Power Plants

Water Chemistry	**pH at 25ºC**	***Comments***
$NH_3 + N_2H_4$	8.5 – 9.6	
N_2H_4 only	7.7 – 8.5	60-100 µg/kg N_2H_4
Chelant + $NH_3 + N_2H_4$		80 µg/kg chelant, 0.8 mg/kg NH_3, 0.2 mg/kg N_2H_4
pH 7 with O_2	6.5 – 7.3	50-200 µg O_2/kg, conductivity <0.1 µS/cm
Combined Mode	8 – 8.5	NH_3+O_2 - NH_3 provides slight pH buffering

IV. CONCLUSION

While there are still many unresolved issues, significant progress has been made in acquiring the data on materials properties needed to enable a short list of candidate alloys to be chosen for longer term testing. Some data on materials properties under SCW conditions are currently available for about 90 alloys. A number of out-reactor test facilities are now operating, and some testing of irradiated materials has also been performed. The planned round robin testing and the databases under development will facilitate comparison of data from different laboratories and enable correlations to be developed (*e.g.*, effect of Cr content of alloys). While the pace is not as rapid, some progress in understanding water chemistry issues such as radiolysis and corrosion product transport in SCW has been made.

Acknowledgements

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References

1. Dollezhal’, N.A., I.Ya. Emel’yanov and P.I. Aleshchenkov, “Development of Power Reactors Such as the Reactor at the Beliy Yar Nuclear Power Plant with Nuclear-superheated Steam”, Report No. 309 at the 3rd Geneva Convention (1964).

2. Wright J.H., and J.F. Paterson, “Status and Application of Supercritical Water Reactors”, *Proc. American Power Conf.*, Vol. XXVIII, p. 139 (1966).

3. Oka, Y., and S. Koshizuka, “Concept and Design of a Supercritical-Pressure Direct Cycle Light Water Reactor”, Nucl. Technology, **103**, 295 (1993).

4. Schulenberg, T., and J. Starflinger, “European Research Project on the High Performance Light Water Reactor”, 4th Int. Symp. on Supercritical Water-Cooled Reactors, Heidelberg, Germany, 2009 March 8-11.

5. Bae, Y.Y., J. Jang, H.Y. Kim, H.-Y. Yoon, H.-O. Kang, K.-M. Bae, “Research Activities on a Supercritical Pressure Water Reactor in Korea”, Nucl. Eng. and Technology, **39**, 273 (2007).

6. Torgerson, D.F., Shalby, B.A., Pang, P., “CANDU Technology for Generation III+ and IV Reactors”, Nucl. Eng. Design **236**, 1565 (2006).

7. A Technology Roadmap for Generation IV Nuclear Energy Systems, Report No. GIF002-00, December 1, (2002) (*http://nuclear.gov*).

8. Feasibility Study of Supercritical Light Water Cooled Reactors for Electric Power Production, Final Report, DE-FG07-02SF22533, INEEL/EXT-04-02539, January, 2005.

9. Chow, C.K., and Khartabil, H.F., “Conceptual Fuel Channel Designs for CANDU-SCWR”, Nuclear Engineering and Technology, **40**, 139 (2007).

10. Kritzer, P., "Corrosion in High-Temperature and Supercritical Water and Aqueous Solutions: A Review", *J. of Supercritical Fluids*, **29**, 1 (2004).

11. Wesolowski, D.J., S.E. Ziemniak, L.M. Anovitz, M.L. Machesky, P. Benezeth, D.A. Palmer, "Solubility and Surface Adsorption Characteristics of Metal Oxides". In: *Aqueous Systems at Elevated Temperatures and Pressures: Physical Chemistry in Water, Steam, and Hydrothermal Solutions* [Palmer *et al.*, (eds.)], Elsevier, Amsterdam, 2004, Chap. 14,

12. Heikinheimo, L., D. Guzonas, and C. Fazio, "Generation IV Materials and Chemistry Research – Common Issues with the SCWR Concept", 4th Int. Symp. On Supercritical Water-Cooled Reactors, Heidelberg, Germany, 2009 March 8-11.

13. Penttilä, S., A. Toivonen, L. Heikinheimo, and R. Novotny, "SCC Properties and Oxidation Behaviour of Austenitic Alloys at Supercritical Water Conditions", 4th International Symposium on Supercritical Water-Cooled Reactors, March 8-11, 2009, Heidelberg, Germany.

14. Was, G.S., P. Ampornrat, G, Gupta, S. Teysseyre, E.A. West, T.R. Allen, K. Sridharan, L. Tan, Y. Chen, X. Ren, C. Pister, "Corrosion and Stress Corrosion Cracking in Supercritical Water", *J. Nucl. Materials,* **371**, 176 (2007).

15. Matsui, H., Y. Sato, N. Saito, F. Kano, K. Ooshima, J. Kaneda, K. Moriya, S. Ohtsuka, Y. Oka "Material Development for Supercritical Water-cooled Reactors", *Proceedings of ICAPP 2007*, Nice, France, May 13-18, 2007.

16. Zheng, W., D. Guzonas and J. Li, "Novel Approach to the Development of In-Core Materials for a Supercritical Water Cooled Reactor", 16 Pacific Basin Nuclear Conference (16 PBNC), Aomori, Japan, Oct 13-18, 2008.

17. Allen, T.R., K. Sridharan, L. Tan, W.E. Windes, J.I. Cole, D.C. Crawford, and G.S. Was, "Materials Challenges for Generation IV Nuclear Energy Systems", Nuclear Technology, **162**, 342 (2008).

18. Cohen, P., (ed.), The ASME Handbook on Water Technology for Thermal Power Systems, Amer. Soc. Mech. Eng., NY, 1989, 1828 pp.

19. Guzonas, D., P. Tremaine, J.-P. Jay-Gerin, "Chemistry Control Challenges in A Supercritical Water Cooled Reactor", International Conference on Water Chemistry of Nuclear Reactor Systems (NPC'08) in Berlin, September 15-18, 2008.

20. Saito, N., Y. Tsuchiya, S. Yamamoto, Y. Akai, T. Yotsuyanagi, M. Domae, Y. Katsumura, "Chemical Thermodynamics Consideration on Corrosion Products in Supercritical Water-Cooled Reactor Coolant", *Nucl. Technol.* **155**, 105 (2006).

21. Kysela, J., V. Švarc, M. Růžičková, "Water Chemistry Specifications For Supercritical Water Cooled Reactors – Possible Options", 4th Int. Symp. On Supercritical Water-Cooled Reactors, Heidelberg, Germany, 2009 March 8-11.

22. Meesungnoen, J., J.-P. Jay-Gerin, and D. Guzonas, "Monte-Carlo Simulations of the Radiolysis of Water at Elevated Temperature and in the Supercritical Regime", Int. Workshop on Supercritical Water Coolant Radiolysis, 14-15 June 2007, Nuclear Research Institute, Rez, Czech Republic.

23. Pommeret, S., “Radiolysis of Supercritical Water: An Experimental and Theoretical Challenge”, Int. Workshop on Supercritical Water Coolant Radiolysis, 14-15 June 2007, Nuclear Research Institute, Rez, Czech-Republic.

24. Katsumura, Y., “Application of Radiation Chemistry to Nuclear Technology”. In: *Charged Particle and Photon Interactions with Matter* [Mozumder, A., Y. Hatano (eds.)], Marcel Dekker, New York, 2004, p. 697.

25. Burrill, K.A., “Water Chemistries and Corrosion Product Transport in Supercritical Water in Reactor Heat Transport Systems”, 8th BNES Conference on Water Chemistry of Nuclear Reactor Systems, Bournemouth, U.K., 22-26 October 2000.

26. Guzonas, D., P. Tremaine, F. Brosseau, “Predicting Activity Transport In A Supercritical Water Cooled Pressure Tube Reactor”, 4th International Symposium on Supercritical Water-Cooled Reactors, Heidelberg, Germany, 2009 March 8-11.

27. Han, Z. and Y. Muroya, “Development of a New Method to Study Elution Properties of Stainless Materials in Subcritical and Supercritical Water”, 4th Int. Symp. On Supercritical Water-Cooled Reactors, Heidelberg, Germany, 2009 March 8-11.

28. Dorsch, T., S. Weiß, “Water Chemistry Regimes for LWRs and Supercritical Fossil Fired Power Plants”, Int. Workshop on Supercritical Water Coolant Radiolysis, Nuclear Research Institute, Rez, Czech Republic, 14-15 June 2007.

29. Růžičková, M., P. Hájek, S. Šmida, R. Všolák, J. Petr, J. Kysela, “Supercritical water loop design for corrosion and water chemistry tests under irradiation”, Nuclear Engineering and Technology, **40**, No. 2. Special issue on the 3rd International Symposium on SCWR, 2007.

LEAD-COOLED FAST REACTOR (LFR): OVERVIEW AND PERSPECTIVES

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I. INTRODUCTION

The *GIF Technology Roadmap* [1] identified the Lead-cooled Fast Reactor (LFR) as a technology with great potential to meet the small-unit electricity needs of remote sites while also offering advantages as a large system for grid-connected power stations. The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides from spent fuel and as a burner/breeder. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant. In the Roadmap, the LFR was primarily envisioned for missions in electricity and hydrogen production, and actinide management.

The application of lead technology to nuclear energy had its start in Russia in the 1970s and 80s where nuclear systems cooled by Lead-Bismuth Eutectic (LBE) were developed and deployed for submarine propulsion. More recently, attention to heavy liquid metal coolants for reactors has developed in several countries around the globe as their advantageous characteristics have gradually become recognized. This paper illustrates the technical progress achieved in the various countries.

II. LFR IN GENERATION IV

International cooperation on LFR within GIF was initiated in October 2004 and the first formal meeting of the Provisional System Steering Committee (LFR-PSSC) was held in March 2005 in Monterey, CA, USA, with participation of representatives from EURATOM, Japan, the United States and experts from the Republic of Korea. Since then, the PSSC has held regular scheduled meetings, roughly twice a year, with additional working sessions to prepare and update the draft LFR System Research Plan (SRP) [2].

The draft SRP was reviewed by the GIF Experts Group (EG) in mid-2007 and again in mid-2008. The formal PSSC meetings were supplemented by additional informal meetings with representatives of the nuclear industry, research organizations and universities involved in LFR development.

The preparation of a System Arrangement for approval by participating GIF members has been considered, but formal agreements are still pending.

The preliminary evaluation of the LFR concepts considered by the PSSC addresses their performance in the areas of sustainability, economics, safety and reliability, proliferation resistance and physical protection.

The designs that are currently proposed as candidates for international cooperation and joint development in the GIF framework are two pool-type reactors:

- the Small Secure Transportable Autonomous Reactor (SSTAR); and
- the European Lead-cooled System (ELSY).

Key design data of SSTAR and ELSY are presented in Table I.

Parameter/system	SSTAR	ELSY
Power (MWe)	19.8	600
Conversion Ratio	~1	~1
Thermal efficiency (%)	44	42
Primary coolant	Lead	Lead
Primary coolant circulation (at power)	Natural	Forced
Primary coolant circulation for DHR	Natural	Natural
Core inlet temperature (°C)	420	400
Core outlet temp. (°C)	567	480
Fuel	Nitrides	MOX, (Nitrides)
Fuel cladding material	Si-Enhanced F/M Stainless Steel	T91 (aluminized)
Peak cladding temp. (°C)	650	550
Fuel pin diameter (mm)	25	10.5
Active core Height/ equivalent diameter (m)	0.976/1.22	0.9/4.32
Primary pumps	-	N° 8, mechanical, integrated in the SG
Working fluid	Supercritical CO_2 at 20MPa, 552°C	Water-superheated steam at 18 MPa, 450°C
Primary/secondary heat transfer system	N°4 Pb-to- CO_2 HXs	N°8 Pb-to-H_2O SGs
Safety grade DHR	Reactor Vessel Air Cooling System + Multiple Direct Reactor Cooling Systems	Reactor Vessel Air Cooling System + Four Direct Reactor Cooling Systems + Four Secondary Loops Systems

TABLE I: Key Design data of GIF LFR concepts

III. SSTAR

The current reference design for the SSTAR [3] in the United States is a 20 MWe natural circulation reactor concept with a small shippable reactor vessel (Figure 1).

The Pb coolant is contained inside a reactor vessel surrounded by a guard vessel. Lead is chosen as the coolant rather than LBE to drastically reduce the amount of alpha-emitting ^{210}Po isotope formed in the coolant relative to LBE, and to eliminate dependency upon bismuth which might be a limited or expensive resource. The Pb flows upward through the core and a chimney above the core formed by a cylindrical shroud. The vessel has a height-to-diameter ratio large enough to facilitate natural circulation heat removal at all power levels up to and exceeding 100% of nominal. The coolant flows through openings near the top of the shroud and enters four modular Pb-to-CO_2 heat exchangers located in the annulus between the reactor vessel and the cylindrical shroud. Inside each heat exchanger, the Pb flows downwards over the exterior of tubes which contain upward-flowing CO_2. The CO_2 enters each heat exchanger through a top entry nozzle which delivers the CO_2 to a lower plenum region. From this lower plenum, the CO_2 enters each of the vertical tubes and flows upward to an upper plenum. The hot CO_2 then exits the heat exchanger through two smaller diameter top entry nozzles. Meanwhile, the Pb exits the heat exchangers and flows downward through the annular downcomer to enter the flow openings in the flow distributor head beneath the core.

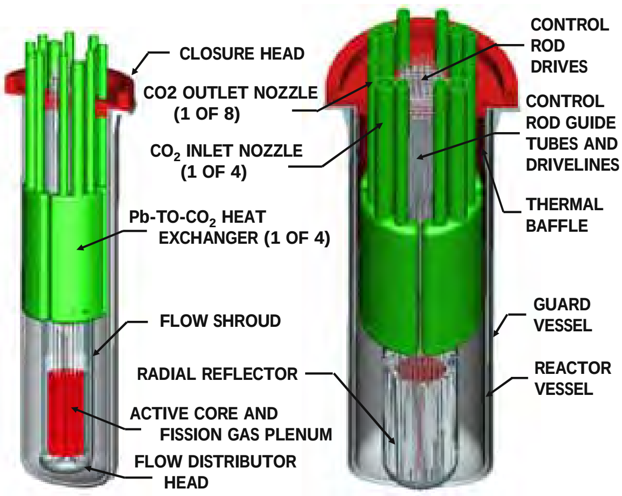

Figure 1: Small Secure Transportable Autonomous Reactor (SSTAR).

Specific features of the lead coolant, the nitride fuel containing transuranic elements, the fast spectrum core, and the small size combine to promote a unique approach to achieve proliferation resistance, while also enabling fissile self-sufficiency, autonomous load following, simplicity of operation, reliability, transportability, as well as a high degree of passive safety.

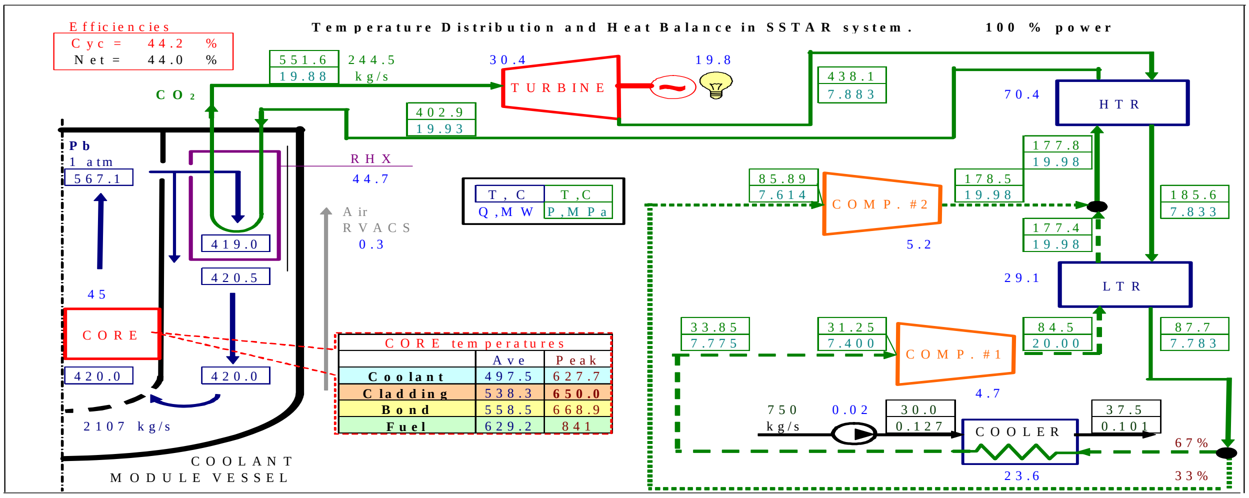

Figure 2: SSTAR pre-conceptual design concept and operating parameters with S-CO_2 Brayton-cycle energy converte

Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished utilizing a supercritical carbon dioxide Brayton cycle power converter (Figure 2).

The SSTAR preconceptual design assumes that a number of advanced technologies will be successfully developed including yet to be developed and code-qualified advanced cladding and structural materials enabling service in Pb for 15 to 30 years at peak cladding temperatures up to about 650°C at a core outlet temperature of 570°C, qualified transuranic nitride fuel meeting fuel performance requirements, whole-core cassette refuelling, and in-service inspection approaches for components immersed in Pb coolant. In turn, the SSTAR concept provides a driver for the development of the advanced technologies. If SSTAR were to be developed for near-term deployment, then the operating system temperatures would likely be reduced (*e.g.*, 480°C, as in the ELSY design) to enable the use of existing codified materials and an existing fuel type such as metallic fuel may have to be qualified and utilized. In the US, an initial scoping investigation has been carried out into the viability of a near-term deployable LFR technology pilot plant/demonstration test reactor (demo) operating at low temperatures enabling the use of existing materials such as T91 ferritic/martensitic steel or Type 316 stainless steel shown in numerous worldwide tests conducted during the past decade to have corrosion resistance to lead alloys at temperatures up to ~ 550°C with active oxygen control. Neutronic and system thermal hydraulic analyses indicate that a 100 MWth lead-cooled metallic-fueled demo with forced flow and a 480°C core outlet temperature supporting the development of both the ELSY and SSTAR LFRs may be a viable concept power converter.

IV. ELSY

The ELSY reference design is a 600 MWe reactor cooled by pure lead [4]. ELSY has been under development since September 2006, and is funded by the Sixth Framework Programme of Euratom. The ELSY project is being conducted by a large consortium of European organizations to demonstrate the possibility of designing a competitive and safe fast critical reactor using simple engineered features, while fully complying with Generation IV goals, including that of minor actinide burning capability.

The use of a compact and simple primary circuit with the additional objective that all internal components be removable, are among the reactor features intended to assure competitive electric energy generation and long-term investment protection. Simplicity is expected to reduce both the capital cost and the construction time; these are also supported by the compactness of the reactor building (*i.e.*, reduced footprint and height). The reduced footprint would be possible due to the elimination of the intermediate cooling system, with a reduced elevation resulting from the design approach of reduced-height components.

One of the main objectives of ELSY from the beginning of the activity has been the identification of innovative solutions to reduce the primary system volume and the complexity of the reactor internals. The result is that most components are unconventional (Figure 3).

A newly designed steam generator (SG), whose volume is about half that of a comparable helical-tube SG, is characterized by a spiral-wound tube bundle. The inlet and outlet ends of each tube are connected to the feed water header and steam header, respectively, both arranged above the reactor roof. An axial-flow primary pump, located inside the inner shell of the SG, provides the head required to force the coolant to enter from the bottom of the SG and to flow in a radial direction. This scheme is almost equivalent to a pure counter-current scheme, because the water circulates in the tube from the outer spirals towards the inner spiral, while the primary coolant flows in the radial direction from the inside to the outside of the SG.

This ensures that the coolant will flow over the SG bundles even in the event of reduction in the primary coolant level in case of leakage from the reactor vessel. As a by-product, the SG unit can be positioned at a higher level in the downcomer and the Reactor Vessel (RV) shortened, accordingly.

All reactor internal structures are removable and in particular the SG Unit can be withdrawn by radial and vertical displacements to disengage the unit from the reactor cover plate.

The core consists of an array of 162 open fuel assemblies (FAs) of square pitch surrounded by reflector-assemblies, a configuration that presents reduced risk of coolant flow blockage. An alternative solution with closed hexagonal FAs has also been retained as a fall-back option. The core is self sufficient in plutonium and can burn its own generated minor actinides with a content at equilibrium of about 1% heavy metal.

The upper part of the FA is peculiar to this novel ELSY design, because it extends well above the fixed reactor cover, and the fuel elements, the weight of which is supported by buoyancy in lead, are fixed at their upper end in the cold gas space, well above the molten lead surface. This avoids the classical problem of a core support grid immersed in the coolant which would require a tricky procedure for In-Service Inspection (ISI) in the molten lead.

FA heads are directly accessible for handling using a simple handling machine that operates in the cover gas at ambient temperature, under full visibility.

Considering the high temperature and other characteristics of the molten lead environment, any approach that foresees the use of in-vessel refuelling equipment, would represent a tremendous R&D effort and substantial associated technical risk, especially because of the need to develop reliable bearings operating in lead, an unknown technology at present. For these reasons the adopted design approach represents a real breakthrough.

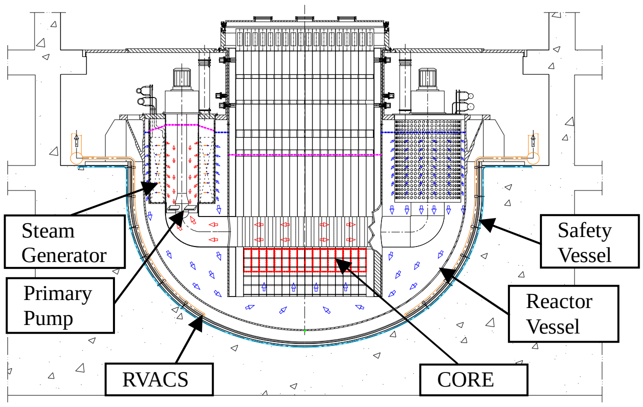

Figure 3: ELSY Primary system configuration.

The installation of SGs inside the reactor vessel is another major challenge of a LFR design that has been resolved by the selected approach. Particular challenges related to the operation of in-vessel SGs include the need for:

- a sensitive and reliable leak detection system;

- a highly reliable depressurization and isolation system.

Careful attention has been also given to the issue of mitigating the consequences of the Steam Generator Tube Rupture (SGTR) accident to reduce the risk of pressurization of the primary boundary; to this end, innovative provisions have been conceived which make the primary system more tolerant of the SGTR event.

The first provision is the elimination of the risk of failure of the water and steam collectors inside the primary boundary by installing them outside the reactor vessel. This approach aims to eliminate by design a potential initiator of a severe accident of low probability but potentially catastrophic consequences.

The second provision is the installation on each tube of a check valve close to the steam header and of a venturi nozzle close to the feed water header.

The third provision aims at ensuring that the flow of any feedwater-steam-primary coolant mixture be re-directed upwards inside the SG, reducing by design the risk of propagation of large pressure waves across the reactor vessel.

This occurs because the inner pressure surge itself promptly causes the closure of the normal radial coolant flow path. The redundant, diverse Decay Heat Removal (DHR) system is provided with (i) steam condensers on the steam loops, (ii) direct reactor cooling loops with innovative lead-water dip coolers using storage water at ambient pressure and (iii) a Reactor Vessel Air Cooling System (RVACS).

General seismic behaviour is strongly improved by the embodied technical solutions, in particular the short-height vessel and the 2D antiseismic supports above the reactor building. Additional loads under investigation are lead sloshing resulting from seismic motion or as a result of a SGTR accident. An extensive safety analysis is also ongoing to address accidents representative of design basis conditions and of design extended conditions.

A preliminary plant Layout showing the main buildings is presented in Figure 4.

Figure 4: ELSY Preliminary plant layout

V. OTHER RELEVANT ACTIVITIES ON LFR

In addition to the ongoing activities in Europe (ELSY) and the USA (SSTAR), it is important also to recognize the ongoing LFR efforts elsewhere.

Research activities in Japan concentrate on the heat transfer performance of LBE in the intermediate loop; two phase flow characteristics of LBE in water and steam; the gas and steam lift performance of LBE; LBE-water direct contact boiling mechanism; the corrosion characteristics and corrosion behaviour of the reactor coolant; the structural and cladding materials; oxygen control with steam injection into LBE; and Polonium behaviour in the coolant system.

The LFR program is strongly promoted in the Center for Research into Innovative Nuclear Energy Systems in Tokyo Institute of Technology. It covers wide areas of lead and LBE coolant studies such as nuclear reactor design studies, cross section measurements, thermal hydraulics experiment especially for steam lift pump, [5] static and dynamic corrosion test, [6] and polonium behavior experiments. [7] The design studies include several kinds of CANDLE reactors [8] and the LBE Cooled Direct Contact Boiling Water Fast Reactor (PBWFR) with electric power of 150 MW. [9]

Two systems are developed in the Republic of Korea, the proliferation-resistant, environment - friendly, accident-tolerant, continual and economical reactor (PEACER) [10] and the BORIS [11]. In the

Russian Federation, two systems are considered: the SVBR-75/100, a LBE-cooled modular fast reactor having a power range of 75 to 100 MWe [12], and the BREST lead-cooled fast reactor concept and the associated fuel cycle. [13]

VI. CONCLUSION

The draft SRP for the Lead-Cooled Fast Reactor has pure lead as the reference coolant and the LBE as a fall-back option. The basic approach recommended in the draft SRP portrays the dual track viability research program with convergence to a single, combined Technology Pilot Plant (TPP) leading to eventual deployment of both types of systems.

The approach adopted aims at addressing the research priorities of each participant party, while developing an integrated and coordinated research program to achieve common objectives and avoid duplication of effort.

Following the successful operation of the TPP around the year 2020,[1] a prototype independent development effort is expected for the central station LFR and the SSTAR.

The design of the industrial prototypes of the central station LFR and of the SSTAR should be planned in such a way as to start construction as soon as beginning of the TPP operation at full power has given assurance of the viability of this new technology.

[1] This is consistent with the European Sustainable Nuclear Energy Technology Platform: "Whilst the SFR remains the reference technology, two alternative technologies for fast reactors, namely the gas-cooled fast reactor (GFR) and the lead-cooled fast reactor (LFR) also need to be assessed at European level. After selection of an alternative technology, an experimental reactor in the range of 50-100 MWth will be needed to gain experience feedback by 2020 on this innovative technology".

Acknowledgements

The work on LFR is supported by the European Commission through the EURATOM Framework Programmes, by the U.S. Department of Energy Generation IV Nuclear Energy System Initiative and by the Tokyo Institute of Technology.

Nomenclature

DHR	Decay Heat Removal
ELSY	European Lead-cooled System
FA	Fuel Assembly
GIF	Generation IV International Forum
LBE	Lead-Bismuth Eutectic
LFR	Lead-Cooled Fast Reactor
ISI	In-Service Inspection
PSSC	Provisional System Steering Committee
RV	Reactor Vessel
RVACS	Reactor Vessel Air Cooling System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SRP	System Research Plan
TPP	Technology Pilot Plant

References

1. GIF-002-00, 2002. “Generation IV Technology Roadmap,” Report GIF-002-00, December 2002.
2. GIF, LFR Provisional System Steering Committee, System Research Plan for the Lead-cooled Fast Reactor (LFR), 2008.
3. Sienicki, J.J., *et al*., “Status of development of the Small Secure Transportable Autonomous reactor (SSTAR) for Worldwide Sustainable Nuclear Energy Supply”, Paper 7218, *Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP)*, Nice, France, 13-18 May, 2007.
4. Cinotti, L., *et al*., “The ELSY Project”, Paper 377, *Proceeding of the International Conference on the Physics of Reactors (PHYSOR)*, Interlaken, Switzerland, 14-19 September, 2008.
5. Takahashi, M., *et al*., “Study on Pb-Bi-Water Direct Contact Two-Phase Flow and Heat Transfer”, Progress in Nuclear Energy, 47. 569-576(2005).
6. Kondo, M., *et al*., “Corrosion of Steels in Lead-Bismuth Flow”, Journal of Nuclear Science and Technology, 43[2] 107-116(2006).
7. Obara, T., *et al*., “Polonium evaporation and adhesion experiments for the development of polonium filter in lead–bismuth cooled reactors”, Progress in Nuclear Energy 50, 556-559(2008).
8. Sekimoto, H., *et al*., “Design study on small CANDLE reactor”, Energy Conversion and Management, 49[7], 1868-1872 (2008).
9. Takahashi M. *et al*., “Pb-Bi-Cooled Direct Contact Boiling Water Small Reactor”, Progress in Nuclear Energy, 47 190-201(2005).
10. Hwang, I.S., “A Sustainable Regional Waste Transmutation System: P E A C E R”, Plenary Invited Paper, ICAPP ‘06, Reno, NV, U.S.A., June 4-6, 2006.
11. Kim, W.J., *et al*., “Supercritical Carbon Dioxide Brayton Power Conversion Cycle Design for Optimized Battery-Type Integral Reactor System”, Paper 6142, *Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP)*, Reno, NV, USA, June 4-8, 2006.
12. Zrodnikov, A.V., *et al*., “Use of Multi-Purpose Modular Fast Reactors SvBR-75/100 in Market Conditions”, Paper 6023, *Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP)*, Reno, Nevada, USA, June 4-8, 2006
13. Adamov, E.O., *et al*., Final report on the ISTC Project #1418: “Naturally Safe Lead-Cooled Fast Reactor for Large Scale Nuclear Power”, Moscow 2001.

LEAD-COOLED FAST REACTOR (LFR) ONGOING R&D AND KEY ISSUES

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I. INTRODUCTION

In 2004, the LFR Provisional System Steering Committee (PSSC) was organized and immediately began their work to develop the LFR System Research Plan (SRP).[1] The committee selected two pool-type reactor concepts as candidates for international cooperation and joint development in the GIF framework: these are the Small Secure Transportable Autonomous Reactor (SSTAR)[2] and the European Lead-cooled System (ELSY).[3]

In evaluating and planning research for these LFR concepts, the LFR-PSSC has followed the general aims of the Generation IV Roadmap;[4] thus, efforts have focused on design optimization with respect to sustainability, economics, safety and reliability, and proliferation resistance and physical protection. Consideration of these factors has guided the identification of research necessary to bring these concepts to fruition.

The needed research activities are identified and described in the SRP. It is expected that in the future, the required efforts could be organized into four major areas of collaboration and formalized as projects. The four areas are: system integration and assessment; lead technology and materials; system and component design; and fuel development.

In this paper, past and ongoing research is summarized and the key technical issues and corresponding future R&D activities are discussed.

II. SUMMARY OF THE KEY ISSUES

Table 1 provides a summary of the key issues for the LFR and the proposed strategy and R&D to address them. Because of the rapid current development of the ELSY system design, the emphasis of this summary is on the research activities and future R&D requirements for the ELSY central station plant. The SSTAR program is proceeding at a slower pace, but shares many of the same research needs and objectives. References 1 and 2 provide additional details of SSTAR-specific requirements and directions. Table 1 and the balance of this paper emphasize the ELSY concept but include issues and directions for both concepts.

III. LEAD TECHNOLOGY AND MATERIALS

Lead is characterized by a high melting point (327.4°C) and a very high boiling point (1 745°C). The high boiling point has a beneficial impact to the safety of the system, whereas the high melting point requires new engineering strategies to prevent freezing of the coolant anywhere in the system, especially at reactor shut down and at refueling. Lead, especially at high temperatures, is also relatively corrosive towards structural materials with a consequent necessity of careful control of lead purity and accurate choice of the structural materials for different components.[5]

General issue	Specific issue	Proposed strategy/needed R&D	Applicability	
			ELSY	SSTAR
Lead technology	Pre-purification.	Verification of industrial capacity to produce high-purity lead.	X	X
	Purification during operation.	Technology for the purification of large quantities of lead to be confirmed.	X	X
	Oxygen control.	Extend oxygen control technology to pure lead for pool reactors.	X	X
Materials resistant to corrosion in lead.	Material corrosion at high temperatures.	Selection of a low core outlet temperature for initial reactor design.	X	
		Development of new materials for service at temperatures up to 650°C		X
	Reactor vessel corrosion.	Vessel temperature limited by design to about 400°C.	X	
		Use of a thermal baffle and Ar-filled annular zone to provide insulating effect to protect reactor vessel		X
	Fuel cladding	Selection of aluminized surface treated steels for cladding	X	
		The use of Si-Enhanced Ferritic/Martensitic Stainless Steel to retard oxidation rate of cladding		X
	Reactor internals	Materials protected by oxygen control	X	X
	Heat removal	Confirmation of the suitability of aluminized steels for steam generator to avoid lead pollution and heat transfer degradation.	X	
		Development of an innovative supercritical CO_2 energy conversion system		X
	Pump impeller*	Test of innovative materials at high lead speed	X	
Potentially high mechanical loading	Earthquake	Reactor building built with 2D seismic isolators + short vessel design.	X	
	SGTR accident	Prevention by design of: - steam entrainment into the core; - reactor vessel pressurization; - pressure wave propagation across the primary system.	X	
	CO2 Tube rupture	safety grade passive pressure relief to vent CO_2, in the event of heat exchanger tube rupture		X
Main safety functions	Diversified, reliable, redundant DHR	Use of both atmospheric air and pool water.	X	
	Diversified, reliable, redundant reactor shut down system	Confirmation of operation of diversified solutions is needed.	X	X
Special operations	Refueling in lead	Innovative solutions are proposed for ELSY. Cassette core replacement design required for SSTAR	X	X
	ISI & Repair	Reduction by design of the need for ISI. Operation of devices at ~400°C in lead needs to be verified.	X	
Fuel and core design	Fuel selection	Nitride fuel in SSTAR and MOX in ELSY for near-term deployment. MA bearing fuel and high burn up fuels to be developed in synergy with SFR.	X	X
	Lead-fuel interaction	To be assessed	X	X
	Failed fuel detection	New solutions to be investigated.	X	X
	Needs of appropriate computer codes.	Qualification of thermal hydraulic and neutronic codes for a LFR.	X	X
Demo	Technology demonstration reactor	Need recognized and requirements definition and initial design studies underway	X	X

TABLE 1: Summary of key issues, proposed strategies and R&D needs

* The pump impeller problem is not an issue with the SSTAR small system because of the use of natural circulation cooling.

During the 1970's and 80's, considerable experience was developed in Russia in the use of Lead-Bismuth Eutectic (LBE) for reactors dedicated to submarine propulsion. Russian researchers have continued to develop new reactor designs based on both LBE (*i.e.*, the SVBR reactor) and lead (*i.e.*, the BREST reactor) as primary coolants.

For the GIF LFR concepts, lead has been chosen as the coolant rather than LBE to drastically reduce the amount of alpha-emitting ^{210}Po isotope formed in the coolant relative to LBE, and to eliminate dependency upon bismuth which might be a limited or expensive resource.

More recently an extensive R&D program was initiated in Europe and is still ongoing. These efforts, conducted under the IP-EUROTRANS, VELLA and ELSY projects of the EURATOM 6th Framework Programme (FP) and of GETMAT of the 7th FP, are addressing many of the main issues identified in Table 1.

In Japan, the Tokyo Institute of Technology is mainly focused on corrosion behaviour of materials and the performance of oxygen sensors in high temperature liquid lead. In addition, recent efforts have been devoted to the development of the LBE reactor concept known as CANDLE.[6] This concept has not to date been included in the LFR SRP and is therefore not discussed further in this summary.

In the USA, in the past considerable effort was devoted to investigations of lead corrosion and materials performance issues as well as system design of the SSTAR reactor, while more recently the focus has included the development of the desired characteristics and design of a technology pilot plant or demonstrator reactor.[2]

III.A. Lead technology

Nuclear grade lead to be used as a coolant in fast reactors is required to be of higher quality than current high-purity industrial lead. It is essential to control the concentrations of impurities, both because of the potential for activation and also because of the possible effect on corrosion, mass transfer and scale formation at heat transfer surfaces.

Contamination of the lead coolant by metal oxide fines is inherent to reactor operations, but will be strictly controlled to minimize this phenomenon. Owing to the fact that reducible metal oxide fines dissolve in the melt with increasing temperature and are therefore desirable for maintaining the amount of dissolved oxygen (buffering effect) and hence the integrity of the oxide barrier against corrosion/erosion, a compromise between extensive purification and effective corrosion protection is being sought and confirmed by testing.

Structural materials will be protected by the superficial oxide barrier generated by the controlled amount of dissolved oxygen in the melt. The theoretical range of dissolved oxygen at which a LFR should be operated is known. Different technologies such as control via cover gas or via treatment of coolant by-pass streams, have been explored over the past several years. The available experience is mainly based on LBE-cooled loop type facilities. The application to pure lead and large pool-type reactors requires additional investigation particularly on determination of oxygen activity level for the chosen thermal cycle, the different technological solutions for oxygen control, the amount and location of the oxygen sensors and the different options for in-service purification.

At present, most of the R&D activities in the area of instrumentation development have been devoted to oxygen sensors; much of the remaining instrumentation is based on equipment that is in conventional use in the nuclear industry, but qualification in the lead environment is needed.

III.B. Structural materials

Corrosion of structural materials in lead is one of the main issues for the design of LFRs.

Experimental campaigns intended to characterize the corrosion behaviour of industrial steels (namely AISI 316 and T91) have been completed.[5]

A larger effort has been dedicated to short/medium term corrosion experiments in stagnant and also in flowing LBE. These studies, which considered coolant flow velocities of 1-2m/s and an exposure time of 2 000 hours were completed at the CORRIDA loop at Forschungszentrum Karlsruhe (FZK), the CU2 loop at the Institute of Physics and Power Engineering (IPPE), the LECOR loop at ENEA, and the LINCE loop at CIEMAT. In addition, a few experiments have been carried out in pure Pb (*i.e.*, CHEOPE III at ENEA). Knowledge is still missing on medium/long term corrosion behaviour in flowing lead. Experiments confirm that corrosion of steels strongly depends on the operating temperature and dissolved oxygen. Indeed, at relatively low oxygen concentration, the corrosion mechanism changes from surface oxidation to dissolution of the structural steel. Moreover, a relationship between oxidation concentration, flow velocity, temperature and stress conditions of the structural material has been observed as well.[7, 8]

Compatibility of ferritic/martensitic and austenitic steels with lead has been extensively studied[5] and it has been demonstrated that generally, in the low temperature range, *e.g.*, below 450°C, and with an adequate oxygen activity in the liquid metal, both types of steels build up an oxide layer which behaves as a corrosion barrier.

However, in the higher temperature range, *i.e.*, above ~500°C, corrosion protection through the oxide barrier seems to fail.[7] Indeed, a mixed corrosion mechanism has been observed, where both metal oxide formation and dissolution of the steel elements occur (Table 2).

Qualification of welding procedures is at an early stage; brazing has not yet been addressed.

It has been demonstrated that, especially in the high temperature range, the corrosion resistance of structural materials can be enhanced by FeAl alloy coating. Corrosion tests performed on GESA treated samples in flowing HLM (heavy liquid metal) up to 600°C have confirmed the effectiveness of this method,[9] but the Al content in the coating needs to be controlled in order to assure a long-term corrosion protection capability. As the next step, composition control, and the development of a qualification method for those surface layers, will be developed. Testing of T91 specimens representative of fuel cladding, FeCrAlY coated and GESA treated (at FZK) will start in 2009 in flowing lead in the CHEOPE loop at ENEA.

<table>
<tr><th>Effective corrosion protection</th><th>Transition zone</th><th>Additional protection needed</th></tr>
<tr><td rowspan="2">Compact stable oxide barrier on ferrite/martensite and austenite</td><td>Oxide formation on ferrite/martensite</td><td>Metal oxide layer unstable</td></tr>
<tr><td>Mixed corrosion mechanism: oxidation/dissolution on austenite</td><td>FeAl alloy coating stable</td></tr>
</table>

400°C 500°C 550°C 600°C

TABLE 2: Protective action via controlled dissolved oxygen at increasing temperature.

T91 and AISI 316 steels have also been tested both in lead and LBE to assess the phenomena of embrittlement and fatigue: the T91-LBE, and certainly the T91-lead combina-tions are subject to embrittlement, while it is still undetermined in the cases of 316L-lead and 316L-LBE. The eventual combined effect of including neutron irradiation has not been sufficiently investigated. A main objective therefore is to determine whether or not irradiation will promote embrittlement and corrosion attack by these heavy liquid metals.

It is expected that the planned post irradiation evaluation (PIE) of the MEGAPIE target will provide unique data regarding the combined effects of irradiation in a proton-neutron spallation environment, corrosion/erosion/embrittlement by flowing LBE and cyclic thermal/mechanical loading on the properties of T91 steel.[10]

Specimens are also being irradiated in a neutron spectrum and in contact with static LBE in the BR2 (at SCK, Belgium) and HFR (at NRG, Netherlands) reactors for exposures up to 5 dpa at temperatures ranging from 300 to

500°C. However, data at higher doses and in a fast neutron spectrum in pure lead are needed for the design of the LFR.

An irradiation campaign of different materials of interest (T91, T91 with treated surfaces and welds and SS316L) has been proposed in the BOR60 reactor (LEXUR II experiment of the GETMAT project) in liquid lead with a maximum exposure of 16 dpa.

It is expected that assessments of fuel cladding and structural core materials, subjected to both high temperature in a lead environment and fast flux, are critical remaining issues.

Near-term deployment of the LFR is possible only by limiting the core outlet temperature to around 500°C. The possibility of operating at higher temperature offered by the high boiling point of lead will be exploited only in the longer term after successful qualification of new materials such as ODS steels, ceramics and refractory metals.

Reactor internals operate at lower temperature than fuel cladding and can be protected by relying on oxide layer formation and oxygen activity control in the melt. An even more favourable condition is seen for the reactor vessel which in normal operating condition can be maintained at a uniform temperature of about 400°C.

With a primary coolant thermal cycle of 400°C-480°C as proposed in ELSY, also the SG tubes operate within an acceptable temperature range, but use of aluminized steels could avoid lead pollution and heat transfer degradation brought about by a thick metal oxide layer.

Because of the relatively high speed between structural material and lead, pump impellers are subjected to severe corrosion-erosion conditions that cannot be sustained in the long term. A new material (Maxthal: Ti_3SiC_2) tested in stagnant conditions with dissolved oxygen and large temperature range has shown remarkably good behaviour. Tests are planned in Europe on specimens exposed to flowing lead at speeds up to 20m/s.

In the case of SSTAR, due to the planned higher operating temperature it has been recognized that additional research is needed for the development and testing of cladding and structural materials for service in Pb at temperatures up to 650°C. One approach that is being considered involves the use of Si-Enhanced Ferritic/Martensitic Stainless Steel to retard the oxidation rate of cladding.

In addition, the design approach to protect the SSTAR reactor vessel against the anticipated elevated lead coolant temperatures incorporates the use of a thermal baffle and Ar-filled annular zone to provide insulating effect.

IV. POTENTIALLY HIGH MECHANICAL LOADING

Peculiar to a LFR design, besides the high density of the coolant, is the integration of the SG or HX equipment inside the reactor vessel. This implies the risk of a large potential load in the case of an earthquake and of a new load brought about by the Steam Generator Tube Rupture (SGTR) or Heat Exchange tube rupture accidents.

IV.A. Earthquake

An ELSY mitigating feature to the effects of the earthquakes is the use of at least 2D seismic isolators which reduce the mechanical loads, but are relatively inefficient against lead sloshing. Qualification of mechanical codes with experimental data is necessary, but no activity has been initiated so far.

IV.B. SG/HX integrated in the reactor vessel

Installation of SGs inside the vessel in a way that enables operation under accident conditions while maintaining a short vessel dimension is a major challenge of the ELSY LFR design.

During reactor operations, the integration of SGs within the vessel requires:

- a sensitive and reliable leak detection system;

- a highly reliable depressurization and isolation system.

In ELSY the feed-water and steam manifolds are arranged above the reactor roof to eliminate the risk of a catastrophic failure inside the primary boundary. Three provisions have been conceived to mitigate the consequences of the SGTR accident.

The first provision is the installation on each tube of a check valve close to the steam header and of a venturi nozzle close to the feed water header.

The second provision aims at ensuring that the flow of any feedwater-steam-primary coolant mixture be re-directed upwards, thereby preventing the risk of large pressure waves propagation across the reactor vessel.

The third provision prevents the pressurization of the vessel by discharging steam into an outer enclosure.

An extensive experimental activity will be carried out to obtain better understanding of each of these phenomena and especially to verify the new solution proposed in ELSY to prevent pressure wave propagation. Preliminary tests are planned in Europe aiming also at qualification of the mechanical codes.

The SSTAR concept relies not on the steam cycle but on a Brayton cycle energy conversion system that is based on supercritical CO_2.[11] In this system, a set of four In-Vessel Pb-to-CO_2 Heat Exchangers operate in which Pb flows downward over the exterior of tubes through which CO_2 flows upward. The reactor system incorporates safety grade passive pressure relief to vent CO_2, in the event of heat exchanger tube rupture. The interest in enhancing plant efficiency with use of the S-CO_2 Brayton cycle has led to goal of operation at a higher coolant temperature, *i.e.* with peak cladding temperatures of up to 650°C. This requirement results in the need for additional materials development.

V. MAIN SAFETY FUNCTIONS

Lead as the coolant requires specific solutions for the two main safety functions of Decay Heat Removal (DHR) and reactor Shut-down.

V.A. Decay heat removal

A small size reactor such as SSTAR can rely on a simple Reactor Vessel Air Cooling System (RVACS) of the type already conceived for the Sodium-cooled Fast Reactor (SFR).

For the larger ELSY system an innovative dip cooler operating with pool water at ambient pressure has been conceived and a mock up will be shortly manufactured for testing in the ICE loop (Integral Circulation Experiment) of the CIRCE facility at Brasimone, Italy.

V.B. Reactor shut down

The design of control rods operating inside a LFR core is at an initial stage and a remaining design effort as well as test qualification remains to be planned. The main issue of concern is control rod insertion time owing to buoyancy.

VI. SPECIAL OPERATIONS

Operations in lead are challenging because of the high temperature, high density and opacity.

VI.A. Refueling in lead

Considering the obvious difficulty of handling fuel elements in lead, special provisions have been adopted both for SSTAR and ELSY to overcome this issue.

The SSTAR small system features a sealed core without refueling or complete cassette core replacement.

For ELSY the fuel elements have been designed with an extended upper part that extends above the lead coolant surface to allow the use of a handling machine operating in gas at ambient temperature.

VI.B. ISI&Repair

Similar issues to those of refueling exist also for In-Service Inspection (ISI). Simplicity of the primary system for both SSTAR and ELSY is one of the keys to address this issue. Thus, the present reference configuration of ELSY with extended fuel elements allows the elimination of the core support plate, one of the most difficult components for ISI. It should also be noted that in ELSY, all in-vessel components are removable for inspection or replacement.

In any case, the capability to perform ISI in lead is an acknowledged issue, and an appropriate R&D program will be initiated.

VII. FUEL AND CORE DESIGN

In general, it is recognized that the LFR and the SFR have considerable overlap in terms of advanced fuels and associated research needs.

To avoid duplication of effort and considering the worldwide limited capability for fuel irradiation, especially in representative fast neutron spectra, fuel development activities for the LFR are mainly devoted to the qualification of fuel cladding, whereas the development of the fuel itself is strongly dependent on the fuel development programme for the SFR.

Peculiar issues requiring research within the LFR programme include the lead-fuel interaction, the detection of failed fuel, and the qualification of advanced fuels (*e.g.* MA-bearing fuels, high-burnup and high-temperature fuels).

The lack of qualified thermal hydraulic and neutronic codes also requires an important R&D effort. A large activity has been already performed to extend to lead the codes qualified for Na and water-cooled reactors. Lead physical data and correlations have been embodied in thermal hydraulic (*e.g.*: Relap, CFD) and neutronic (*e.g.* ERANOS, FLUKA, MCNP) codes.

In particular the data resulting from the MEGAPIE irradiation test and post-test analyses is valuable for both thermal hydraulics and neutronics.

Qualification of neutronic codes is also planned in the GUINEVERE project: a lead-based, zero-power test facility is being assembled at SCK-CEN in close collaboration with several European Partners in “IP-EUROTRANS”.

The GUINEVERE-project will provide a unique experiment with a continuous beam coupled to a fast-spectrum, sub-critical reactor allowing full investigation of the methodology of reactivity monitoring for subcritical cores, but also offering possibilities for zero-power critical experiments with a pure lead-cooled core.

Several studies have shown that the standard models used in current computational fluid dynamic (CFD) codes are not sufficient to predict adequately heat transfer in heavy metal environment.

A thorough understanding of the thermal hydraulic behaviour of complex components in a pool-type reactor will be gained by three different experiments, which have the aim to characterize, respectively, a single fuel rod, a representative fuel bundle, and a cooling loop of a core sector.

(i) In the single rod experiment at the TALL facility (KTH, Sweden), a pin made of T91 has been tested with 3-21 kW input power range and coolant flow speed from 0.3 m/s for natural convection and up to 2.3 m/s for forced convection.

(ii) A Mock up of a fuel rod bundle with 19 rods, 430 kW, is in assembly (at FZK, Germany), redundantly equipped with instrumentation to measure local temperatures and flow rate distribution within the sub-channels.

(iii) The mock up of a 800 kW, 37 rods fuel rod bundle is under procurement to be installed in the ICE loop of the CIRCE facility (at ENEA, Italy). The ICE loop is representative of a typical pool configu–ration with a small riser and a large downcomer. Operation in forced and natural circulation can be simulated as well as the transient behavior from forced to natural circulation and the phenomenon of lead stratification in the downcomer.

VIII. CONCLUSION

The LFR systems under consideration offer great promise in terms of the potential for providing cost effective, simple and robust fast reactor concepts that are essential to long-term sustainability of the nuclear energy option.

Recent efforts, particularly in the development of the ELSY concept, have gone a long way toward verifying the advantages of lead cooled systems. Clearly additional work needs to be done, but overall, the prospects continue to appear very positive.

The SRP lays out a dual track approach to completing a cooperative research programme for the two recommended systems with convergence to the design of a single, combined Technology Pilot Plant (TPP) to support the eventual deployment of both types of systems. A focus on the design of a TPP suitable to meet the demonstration and research needs of the small as well as central station LFR concepts is an important adjunct to the completion of the research described in the SRP and summarized in this paper.

Acknowledgements

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Nomenclature

BOR60	Sodium-cooled research reactor at the Russian Scientific Research and Design Institute (NIKIET)
BR2	Belgian Reactor-2
CANDLE	Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy
CHEOPE	CHEmistry OPErations facility at ENEA, Brasimone, Italy.
CIRCE	CIRcolazione Eutettico facility at ENEA, Brasimone, Italy
DHR	Decay Heat Removal
ELSY	European Lead-cooled System
GUINEVERE	Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor, facility at SCK-CEN, Belgium
GESA	Gepulste Elektronen-Strahl Anlage, method for surface treatment
HFR	High Flux Reactor at the Joint Research Center (JRC) in Petten
HX	Heat Exchanger
ISI	In-Service Inspection
LBE	Lead-Bismuth Eutectic
LFR	Lead-cooled Fast Reactor
MA	Minor Actinide
MEGAPIE	Experiment to demonstrate a liquid metal spallation target at the Paul Scherrer Institut
MOX	Mixed Oxide
PSSC	Provisional System Steering Committee
RVACS	Reactor Vessel Air Cooling System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SFR	Sodium-cooled Fast Reactor
SRP	System Research Plan
SSTAR	Small Secure Transportable Autonomous Reactor

TPP	Technology Pilot Plant
VELLA	Virtual European Lead Laboratory

References

1. GIF LFR Provisional System Steering Committee, System Research Plan for the Lead-cooled Fast Reactor (LFR), 2008.

2. Smith, C., W. Halsey, N. Brown, J. Sienicki, A. Moisseytsev, D. Wade, "SSTAR: The US lead-cooled fast reactor (LFR)," *Journal of Nuclear Materials*, Volume 376, Issue 3, 15 June 2008, pp. 255-259.

3. Cinotti, L., G. Locatelli, H. Aït Abderrahim, S. Monti, G. Benamati, K. Tucek, D. Struwe, A. Orden, G. Corsini, D. Le Carpentier, "The ELSY Project", Paper 377, Proceedings of the International Conference on the Physics of Reactors (PHYSOR), Interlaken, Switzerland, 14-19 September, 2008.

4. GIF-002-00, 2002. "Geneneration IV Technology Roadmap," Report GIF-002-00, December 2002.

5. Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies, OECD-NEA 6195, 2007.

6. Sekimoto, H., A. Nagata, "Fuel Cycle for "CANDLE" Reactors," Proc. of Workshop on Advanced Reactors With Innovative Fuels ARWIF-2008, Tsuruga/Fukui, 20-22 February 2008.

7. Fazio, C., A. Alamo, A. Almazouzi, D. Gomez-Briceno, F. Groeschel, F. Roelofs, P. Turroni, and J.U. Knebel, "Assessment of Reference structural materials, heavy liquid metal technology and thermal-hydraulics for European waste transmutation ADS", Proceedings of GLOBAL 2005 Tsukuba, Japan, Oct 9-13, 2005.

8. Fazio, C., A. Alamo, J. Henry, A. Almazouzi, D. Gomez-Briceno, L. Soler, J.B. Vogt, F. Groeschel, F. Roelofs, P. Turroni, R. Stieglitz and J.U. Knebel, "European research on heavy liquid metal technology for advanced reactor systems", 15th Pacific Bacin Nuclear Conference, 2006, Sydney.

9. Fazio, C., A. Almazouzi, F. Groeschel, J. Henry, F. Roelofs, L. Soler Crespo, P. Turroni and J. Knebel, "Structural materials, thermal-hydraulics studies and development of heavy liquid metal technologies for Accelerator Driven sub-critical transmutation Systems", Proceedings of the Jahrestagung Kerntechnik 2008, May 27-29, 2008 Hamburg, Germany.

10. Fazio, C., *et al*., "European cross-cutting research on structural materials for Generation IV and transmutation systems," J. Nucl. Mater. (2009), doi:10.1016/j.jnucmat.2009.03.020.

11. Sienicki, J.J., *et al*., "Supercritical Carbon Dioxide Brayton Cycle Energy Conversion for Sodium-Cooled Fast Reactors/Fast Burner Reactors," GLOBAL 2007, Advanced Nuclear Fuel Cycles and Systems, Boise, September 9-13, 2007.

THE MOLTEN SALT REACTOR (MSR) IN GENERATION IV: OVERVIEW AND PERSPECTIVES

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I. INTRODUCTION

In a Molten Salt Reactor (MSR), the fuel is dissolved in a fluoride salt coolant. The technology was partly developed, including two demonstration reactors, in the 1950's and 1960's in USA (ORNL). Compared with solid-fuelled reactors, MSR systems have lower fissile inventories, are insensitive to fuel radiation damage that can limit fissile and fertile material utilization, provide the possibility of continuous fission-product removal, avoid the expense of fabricating fuel elements, give the possibility of adding makeup fuel as needed, which precludes the need for providing excess reactivity, and employ 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.

Prior MSRs were mainly considered as thermal-neutron-spectrum graphite-moderated concepts. Since 2005, R&D has focused on the development of fast-spectrum MSR concepts (MSFR) combining the generic assets of fast neutron reactors (extended resource utilization, waste minimization) to those relating to molten salt fluorides as fluid fuel and coolant (favourable thermal-hydraulic properties, high boiling temperature, optical transparency). In addition, MSFR exhibit large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors. [4-8] MSFR has been recognized as a long term alternative to solid-fuelled fast neutron systems with unique potential (negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle…).

Apart from MSR systems, other advanced reactor concepts are being studied, which use the liquid salt technology, as a primary coolant for the Advanced High-Temperature Reactor (AHTR)[11] or intermediate coolant, as an alternative to secondary sodium, for Sodium Fast Reactors (SFR) and to intermediate helium for Very High Temperature Reactors (VHTR).

More generally, the development of higher temperature salts as coolants would open new nuclear and non-nuclear applications. These salts could also facilitate heat transfer for nuclear hydrogen production concepts, concentrated solar electricity generation, oil refineries, and shale oil processing facilities amongst other applications. [3]

In brief, there has been a significant renewal of interests for liquid salt applications.

The paper shows the main technical progress achieved in the countries participating to the R&D effort on the MSR in GIF and remaining issues to be addressed.

II. MSR IN GENERATION IV

A decision to establish a Provisional System Steering Committee (PSSC) 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 PSSC. Russia has played an important role in identifying R&D issues basing on long-lasting R&D programs initiated the 1970s.

The renewal and diversification of interests in molten salts have led the MSR PSSC to a shift of the R&D orientations and objectives initially promoted in the original Generation IV Roadmap issued in 2002, [1] in order to encompass in a consistent body the different applications envisioned today for fuel and coolant salts. [2]

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. The potential of MSFR has been assessed but specific technological challenges must be addressed and the safety approach has to be established.
- The Advanced High Temperature Reactor (AHTR) is a high temperature reactor with higher power density than the VHTR and passive safety potential from small to very high unit power (> 2 400 MWt).

In Russia, the efficiency of MSR for actinide burning has been investigated. This resulted into the single stream Li, Na,Be/F MOlten Salt Actinide Recycler & Transmuter (MOSART) fast spectrum system fuelled with compositions of plutonium plus minor actinide trifluorides (AnF_3) from UOX and MOX LWR spent fuel without U-Th support. [13]

In addition, the opportunities offered by liquid salts for intermediate heat transport in other systems (SFR, LFR, VHTR) are being investigated. Liquid salts offer two potential advantages: smaller equipment size because of the higher volumetric heat capacity of the salts; and no gross chemical exothermal reactions between the reactor, intermediate loop, and power cycle coolants.

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

The factorization into projects in the SRP emphasizes cross-cutting R&D areas. A major commonality is the understanding and mastering of fuel and coolant salts technologies, including development of structural materials, reliable knowledge on physical properties for fuel and coolant salts, fuel and coolant salts clean-up, chemical and analytical R&D for fuel and coolant behaviour.

III. MSFR REFERENCE OPTIONS

Starting from the ORNL Molten Salt Breeder Reactor project (MSBR), an innovative concept has been proposed [4, 5], resulting from extensive parametric studies in which various core arrangements, reprocessing performances and salt compositions were investigated. The primary feature of the MSFR (Molten Salt Fast Reactor) concept is the removal of the graphite moderator from the core (graphite-free core).

In terms of fuel cycle, two basic options have been investigated, ^{233}U-started MSFR and TRU-started MSFR.

Realistic drawings showing the main MSFR components and their arrangement in the vessel have been elaborated. Figure 1 displays a schematic drawing of a vertical section of the MSFR while Table 1 presents some characteristics of the reactor.

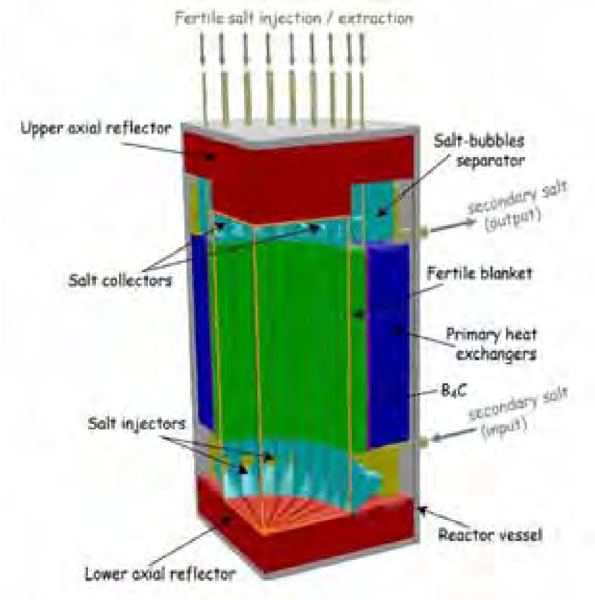

Figure 1: Schematic view of a quarter of the MSFR

The core is a single cylinder (diameter equal to height) where nuclear reactions take place within the flowing fuel salt. It is made of three volumes: the active core, the upper plenum and the lower plenum. The fuel salt is a binary salt, composed of LiF enriched in ^{7}Li (99.999%) and heavy nuclei (HN) amongst which the fissile element, ^{233}U or Pu. The $(HN)F_4$ proportion is set at 22.5 mol% (eutectic point), corresponding to a melting temperature of 550°C. The choice of this fuel salt composition relies on many systematic studies (influence of chemical reprocessing on neutronic behavior, burning capabilities, deterministic safety level, deployment capabilities). [6-10] This salt composition leads to a fast neutron spectrum in the core. The outer core structures and heat exchangers are protected by thick reflectors designed to absorb more than 80% of the escaping neutron flux. These reflectors are themselves surrounded by a 10 cm thick neutronic protection of B_4C absorbing remaining neutrons. Axial reflectors are made of nickel-based alloys. The radial reflector consists of a fertile blanket (50 cm thick) filled with a fertile salt of LiF-ThF_4 with 22.5 mol% ^{232}Th.

The level of deterministic safety reached by the concept is excellent since the feedback coefficients of the MSFR are negative in both ^{233}U and TRU starting modes. [6,8,10] The total feedback coefficient is equal to -6 pcm/°C when the equilibrium state of the reactor has been reached and the density coefficient, which for MSRs can also be viewed as a void coefficient, is also largely negative at about -3 pcm/°C.

Thermal power (MWt)	3000				
Fuel molten salt composition (mol%)	LiF-ThF_4-$^{233}UF_4$ or LiF-ThF_4-$(Pu\text{-}MA)F_3$ with LiF = 77.5 mol%				
Fertile blanket molten salt composition (mol%)	LiF-ThF_4 (77.5-22.5)				
Melting point (°C)	550				
Operating temperature (°C)	700-800				
Initial inventory (kg)	^{233}U-started MSFR		TRU-started MSFR		
	Th	^{233}U	Th	Actinide	
	38300	5060	30600	Pu	11200
				Np	800
				Am	680
				Cm	115
Density (g/cm^3)	4.1				
Dilatation coefficient (/°C)	10^{-3}				
Core dimensions (m)	Radius: 1.15 Height: 2.30				
Fuel salt volume (m^3)	18 9 out of the core 9 in the core				
Blanket salt volume (m^3)	8				
Thorium consumption (ton/year)	1.112				
^{233}U production (kg/year)	93 (^{233}U-started MSFR) 188 during 20 years then 93 (TRU-started MSFR)				
Breeding ratio (^{233}U-started MSFR)	1.085				

Table 1: Reference design characteristics of the MSFR

A good indicator of the deployment capability is the doubling time, defined by the operation time leading to the ^{233}U inventory of a new reactor of the same type through breeding. For a ^{233}U-MSFR, the annual ^{233}U production is 120 kg which corresponds to 50 years doubling time per reactor. [6, 9] Starting a MSFR from Generation II or III reactors spent fuel is more favourable and yields 35 years doubling time. Indeed, the presence of other fissile elements decreases the consumption of ^{233}U and improves the deployment capability of the concept.

IV. AHTR REFERENCE PLANT CONCEPT

The defining aspects of an Advanced High Temperature Reactor (AHTR) are the use of coated particle fuel embedded within a graphitic matrix cooled by liquid fluoride salt. [11] A Pebble Bed Advanced High Temperature Reactor (PB-AHTR) operating at ~900 MWt is the most actively developing commercial scale plant design. [12] The plant design is currently transitioning from a primarily conceptual to an initial engineering scoping phase. A half cross section of the core concept is shown in Figure 2.

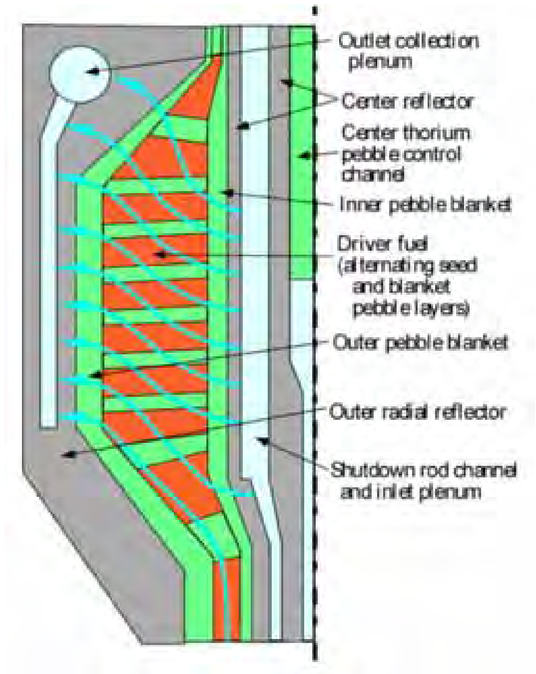

Figure 2: Half Cross Section of PB-AHTR Core

A major design refinement of the current core is the use of inner and outer pebble blankets to reduce the radiation damage to the fixed reflector graphite. The power density of a salt cooled pebble bed is 4-8x greater than that of its gas-cooled cousin. The resultant higher flux level would necessitate more frequent reflector graphite replacement without the use of blanket pebble layer. The controlled motion of a structured pebble assembly has recently been demonstrated using simulant materials at U.C. Berkeley, along with friction coefficient measurements for graphite pebbles verifying that fluoride salts act as effective lubricants and that friction coefficients are very close to those for the simulant materials. Pebble motion demonstration using prototypic materials and temperatures will be a key aspect of future R&D on the PB-AHTR.

V. R&D PROGRESS AND REMAINING ISSUES IN SPECIFIC AREAS

Significant progress has been achieved in 2008 in critical areas of MSR-AHTR R&D. In brief, the essential facts are the following:

1. Salt selection for different applications is stabilized, the needs of complementary data have been clarified. [14, 18]
2. A strongly improved (versus MSBR) fuel salt clean-up scheme has been developed. [8, 15, 16]
3. Criticality tests are being performed for the assessment of MSR and AHTR fuel and core behaviour.

Those topics are the subject of the following sub-sections.

Although progress has been made in the area, the assessment of structural materials remains challenging for MSFR and AHTR as both concepts are supposed to operate at temperatures higher compared to MSBR.

V.A Salt selection for different applications

Potential salt systems have been critically reviewed in the frame of the ALISIA project in the EURATOM 6th FWP. [14] Reference compo–sitions have been proposed or confirmed (Table 2).

Reactor type	Neutron spectrum	Application	Carrier salt	Fuel system
MSR-Breeder	Thermal	Fuel	$^{7}LiF-BeF_2$	$^{7}LiF-BeF_2-ThF_4-UF_4$
	Non-moderated	Fuel	$^{7}LiF-ThF_4$	$^{7}LiF-ThF_4-UF_4$
				$^{7}LiF-ThF_4-PuF_3$
MSR-Breeder	T/NM	Secondary coolant	$NaF-NaBF_4$	
MSR-Burner	Fast	Fuel	LiF-NaF	$LiF-(NaF)-AnF_4-AnF_3$
			$LiF-(NaF)-BeF_2$	$LiF-(NaF)-BeF_2-AnF_4-AnF_3$
			$LiF-NaF-ThF_4$	
AHTR	Thermal	Primary coolant	$^{7}LiF-BeF_2$	
SFR		Intermediate coolant	$NaNO_3-KNO_3-(NaNO_2)$	

Table 2: Fuel and coolant salts for different applications

The $^{7}LiF-BeF_2$ (66:34 in mol%) salt is the selected fuel carrier for the moderated (thermal) molten salt thorium breeder, giving as fuel salt $^{7}LiF-BeF_2-ThF_4-UF_4$. From neutronic as well as chemical point of view, there are no alternatives for this salt that do not penalise the breeding capacity of the reactor.

$^{7}LiF-ThF_4$ (78:22 or even 71:29 in mol%) is the reference fuel solvent composition for the fast spectrum molten salt thorium breeder reactor (MSFR). The neutronic analysis of the MSFR concept has demonstrated the feasibility of the concept, but it must still be clarified whether the physico-chemical properties (melting temperatures, solubility for the actinide trifluorides, density, expansivity, viscosity, thermal conductivity, heat capacity) of this salt fuelled by significant amount of UF_4 (2-4% of the total heavy nuclei in the moderated and 12-18% in the fast systems) or AnF_3 (up to 25% of the total heavy nuclei in the fast concept) are consistent with safe operation of the reactor and fuel salt clean-up unit. To tune these properties, addition of other components is possible. The most obvious is BeF_2 but there is an incentive to keep the content of this material low (*e.g.* $71LiF-2BeF_2-27ThF_4$ or $75LiF-5BeF_2-20ThF_4$ in mol%) or even zero. Alternatives are NaF and possibly CaF_2. Therefore, the $^{7}LiF-NaF-ThF_4$ system must be further analysed, whereas scoping studies of the $^{7}LiF-CaF_2-ThF_4$ system are required, to assess the pros and cons for both molten salt mixtures, including suitability for fuel salt processing.

The molten salt actinide burner is a fast spectrum concept too. The carrier salt for this application must have good solubility for the actinide trifluorides and this can be achieved using $^{7}LiF-NaF-(KF)$ as solvent or $^{7}LiF-(NaF)-BeF_2$ melt. Again, the goal is to keep the content of BeF_2 low or even zero. An interesting alternative is the use of plutonium and minor actinides as start-up for the thorium cycle in the MSR, leading to $^{7}LiF-NaF-ThF_4$ carrier salt.

In summary, it is clear that the $^{7}LiF-(NaF)-AnF_4-AnF_3$ salt (where An represent actinides) is the key system to be further investigated in parallel to the $^{7}LiF-(NaF)-BeF_2-AnF_4-AnF_3$ system. Optimisation of the fractions of the components is still needed with respect to mentioned-above physico-chemical properties, corrosion behavior in the Ni-Mo alloys and fuel salt processing.

For coolant salts, one has to make a distinction between salt for in-core use (primary coolant) and salts for out-of-core use (secondary or intermediate coolants). For primary coolants in thermal reactors, the requirements are very similar to thermal breeder reactors and ^{7}LiF-BeF$_2$ (66-34 with Tm=458°C) is the main candidate, with ^{7}LiF-NaF-KF (46-11.5-42.5 with Tm=454°C), LiF-NaF-RbF (46.5-6.5-47 with Tm=426°C) and ^{7}LiF-NaF-BeF$_2$ (30.5-31-38.5 with Tm=316°C) as alternatives. Note that the last alternative molten salt mixture has the lowest liquidus temperature.

For secondary coolant applications, neither neutronic considerations nor actinide solubility play a role and a wider choice of materials is possible. For MSRs in which tritium control is the main concern, the NaF-NaBF$_4$ (8:92 with Tm=385°C) system is the prime candidate, mainly because of its satisfactory tritium trapping. A ternary salt LiF-NaF-BeF$_2$ should be considered in future studies as alternative secondary coolant because a freezing temperature range of about 315-335°C would be a practical value for engineering consideration. Because closed gas Brayton cycles can mitigate both the tritium and the melting point concerns, LiF-NaF-KF or NaCl-KCl-MgCl$_2$ may also be considered as a secondary salt.

Finally, heat transfer for lower temperature applications (below 600°C) requires a cheap and stable salt. NaNO$_3$-KNO$_3$ possibly with addition of NaNO$_2$ is the main candidate identified at this stage.

V.B Fuel salt clean-up scheme

The salt processing scheme relies on both on-line and batch processes to satisfy the constraints for a smooth reactor operation while minimizing losses to waste streams. ORNL experiments have provided some data mainly for the on-line gaseous fission product extraction process.

Acquisition of fundamental data for the separation processes is needed especially for the actinide-lanthanide separation. The extraction of lanthanides has to be done because of the low solubility of these trifluoride elements and neutronic captures that decrease the reactivity balance.

The progress made in core design in the last two years has opened the door for the definition of an improved fuel salt reprocessing scheme with a realistic fuel clean-up rate (40 l/day) and minimized losses to wastes. [6,8,15]

The proposed reference processing scheme is shown in Figure 3. The first step (green box) involves an on-line gaseous extraction with

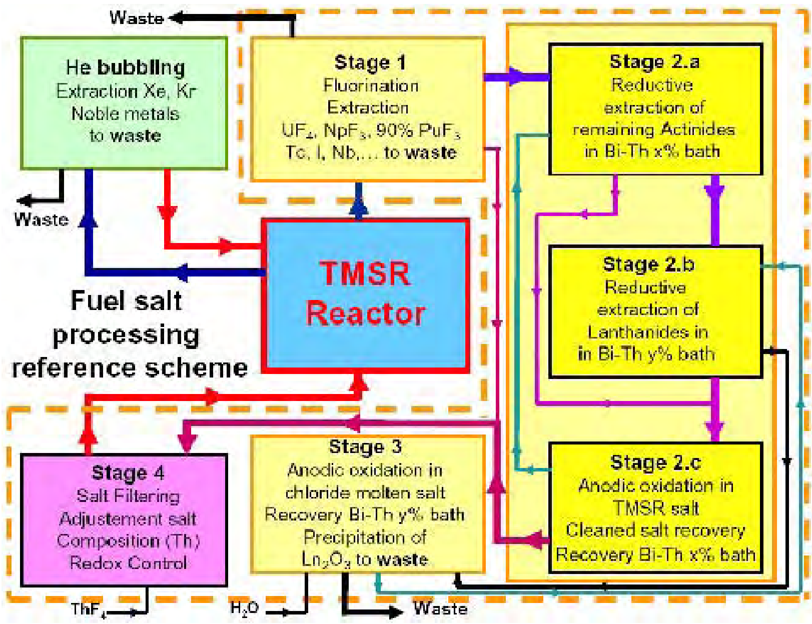

Figure 3: TMSR (MSFR) reference fuel salt processing

helium bubbling to remove gaseous fission products, Xe and Kr, and a part of the noble metals from fuel circuit. On the other hand, a batch fuel process separates the actinides which are returned to the reactor salt from the harmful fission products (mostly lanthanides). The fuel clean-up rate has been set at 40 liters per day, corresponding to the processing of 100 kg heavy nuclei per day. This value is almost two orders of magnitude less than the reference MSBR scheme.

The reference scheme depicted in Figure 3 involves 4 stages for the batch on-site fuel processing. The peculiarity of the concept appears in stages 2 and 3 by combining chemical and electrochemical methods for the extraction and the back extraction of actinides and lanthanides. This choice leads to fuel processing without effluent volume variation and the fuel processing balance is reduced to only one reaction: $2LnF_3 + 3H_2O(g) = Ln_2O_3 + 6HF(g)$.

Critical steps of the new fuel clean-up scheme are addressed and will be experimentally assessed in new facilities. The design and construction of a molten salt loop to study both He bubbling efficiency and material corrosion attack has been initiated. An efficient technique for actinide/lanthanide separation is under qualification. [16]

V.C Criticality tests for the assessment of MSR and AHTR fuel and core behaviour

The SPHINX (SPent Hot fuel Incinerator by Neutron fluX) project was originally defined as a suitable experimental basis at representative scale for the demonstration of MSR-burner feasibility. [17] It relies on the utilization of the zero power experimental reactor LR-0 being operated in the Nuclear Research Institute Řež (NRI), Czech Republic. This full-scale physical model of the PWR cores was modified in order to allow the measurement of all the neutronic characteristics of the MSR burner and/or breeder blanket, at first by room temperature and in future stage by conditions close to operational. (Figure 4).

Figure 4: LR-0 zero power critical test facility

Because two baseline concepts (MSFR, AHTR) are now considered in Generation IV, a corresponding broadening of the SPHINX project was discussed and formally adopted at the end of 2008. The LR-0 will thus be used for the validation of AHTR neutronics models (reactivity coefficients…) in the frame of a collaboration between the Czech Republic (NRI) and USA (University of California, Berkeley).

Two versions of EROS elementary blocks, as simplified models of the AHTR core module, have been designed and manufactured. During December 2008, the critical tests of both those elementary blocks were performed. The simplified models are completely ready for complex testing of experimental and measuring methods for detailed neutron field distribution and principle neutronic characteristics prediction.

VI. CONCLUSION

Europe (Euratom), France and USA participate in the Generation IV MSR Steering Committee. Although the European and USA interests are focused on different baseline concepts (MSFR and AHTR, respectively), large commonalities in basic R&D areas (liquid salt technology, materials) exist and the Generation IV framework is useful to optimize the R&D effort.

In USA, a PB-AHTR (900 MWt) is being developed most actively. A research, development and demonstration roadmap is under study for component testing to support a PB-AHTR prototype scale plant and a development path for the structural materials is being established.

In Europe, since 2005, R&D on MSR has been focused on fast spectrum concepts (MSFR) which have been recognized as long term alternatives to solid-fuelled fast neutron reactors with attractive features (very negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle…). MSFR designs are available for breeding and for minor actinide burning. They are robust reference configurations (with significant improvement compared to MSBR), allowing to concentrate on specific R&D issues [19].

A network on MSR R&D has been active in Europe from 2001 to 2008 with financial support by EURATOM. In parallel, ISTC has provided another efficient way of collaboration between Russian research organizations, European partners and non-European partners (USA, Canada, IAEA).

The GIF plays an important role to enhance and harmonize international collabo–ration on the R&D conducted in the different contexts.

Acknowledgements

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Nomenclature

AHTR	Advanced High Temperature Reactor
An	Actinide
GIF	Generation IV International Forum
LWR	Light Water Reactor
MSR	Molten Salt Reactor
MSFR	Molten Salt Fast Reactor
PB-AHTR	Pebble Bed Advanced High Temperature Reactor
PSSC	Provisional System Steering Committee
SRP	System Research Plan
UOX	Uranium Oxide

References

1. GIF-002-00, 2002. “Generation IV Technology Roadmap,” Report GIF-002-00, December 2002.
2. GIF, MSR Provisional System Steering Committee, System Research Plan for the Molten Salt Reactor (MSR), 2008.
3. Forsberg, C.W., C. Renault, C. Le Brun, E. Merle-Lucotte, V. Ignatiev, “Liquid Salt Applications and Molten Salt Reactors”, *Revue Générale du Nucléaire* (RGN) N° 4/2007, 63 (2007).
4. Nuttin, A., D. Heuer, *et al.*, “Potential of Thorium Molten Salt Reactors”, Prog. in Nucl. En., 46, 77-99 (2005).
5. Mathieu, L., D. Heuer, *et al*, “The Thorium Molten Salt Reactor: Moving on from the MSBR”, Prog. in Nucl. En., 48, 664-679 (2006).

6. Merle-Lucotte, E., D. Heuer, *et al.*, "Optimization and simplification of the concept of non-moderated Thorium Molten Salt Reactor", Proceedings of the International Conference on the Physics of Reactors, PHYSOR 2008, Interlaken, Switzerland (2008).

7. Merle-Lucotte, E., D. Heuer, C. Le Brun; and J.M. Loiseaux, "Scenarios for a Worldwide Deployment of Nuclear Power", *International Journal of Nuclear Governance*, Economy and Ecology, 1, Issue 2, 168-192 (2006).

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. Merle-Lucotte, E., D. Heuer, M. Allibert, X. Doligez, V. Ghetta, "Minimizing the Fissile Inventory of the Molten Salt Fast Reactor", Proceedings of the Advances in Nuclear Fuel Management IV (ANFM 2009) International Conference, Hilton Head Island, USA (2009).

10. Merle-Lucotte, E., D. Heuer, M. Allibert, X. Doligez, V. Ghetta, "Optimizing the Burning Efficiency and the Deployment Capacities of the Molten Salt Fast Reactor", Contribution 9149, Proceedings of the GLOBAL 2009 International Conference, Paris, France (2009).

11. 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.

12. Bardet, Ph., *et al.*, "Design, Analysis and Development of the Modular PB-AHTR", Proc. 2008 International Congress on Advances in Nuclear Power Plants (ICAPP'08), Anaheim, CA USA, June 8-12, 2008.

13. Ignatiev V., O. Feynberg, R. Zakirovn *et al.*, "Characteristics of MOlten Salt Actinide Recycler and Transmuter system", Proc of ICENES-2005, Brussels, Belgium, August 21- 26, 2005.

14. 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 6th FWP, 2008.

15. Delpech, S., G. Picard, "Optimization of fuel reprocessing scheme for innovative molten salt reactor", Molten Salts Joint Symposium, Kobe, Japan, October 2008.

16. Delpech, S., *et al.*, "Actinides/lanthanides separation for the Thorium Molten Salt Reactor fuel treatment", ATALANTE 2008, Montpellier, France (2008).

17. 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.

18. 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.

19. Renault, C., *et al*., "The Molten Salt Reactor (MSR) – R&D Status and Perspectives in Europe", FISA 2009, Prague, June 22-24, 2009.

MSFR: MATERIAL ISSUES AND THE EFFECT OF CHEMISTRY CONTROL

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I INTRODUCTION

An innovative molten salt reactor concept, the MSFR (Molten Salt Fast Reactor) is developed by CNRS (France) since 2004. [1,2] Based on the particularity of using a liquid fuel, this concept is derived from the American molten salt reactors (included the demonstrator MSRE) developed in the 1960s. [3-5]

In MSFR, the ORNL (Oak Ridge National Laboratory) MSBR concept has been revisited by removing graphite and BeF_2. The neutron spectrum is fast and the reprocessing rate strongly reduced down to 40 liters per day (compared to 4000l/day in the MSBR concept) to get a positive breeding gain. The reactor is started with ^{233}U or with a Pu and minor actinides (MA) mixture from PWR spent fuel. The MA consumption with burn-up demonstrates the burner capability of MSFR. [1,2]

The structural materials retained for MSR container are Ni-based alloys with a low concentration of Cr. The composition of Hastelloy N (Ni-Mo-Cr system) optimized by ORNL researchers is already a good candidate for temperature up to 750°C. The operating temperatures chosen in neutronic calculations of MSFR systems are ranged between 700 and 850°C. For this high temperature domain, the replacement of Mo by W looks promising from the mechanical properties point of view.

This paper addresses the issue of structural materials considering the mechanical properties at high temperature, the neutronic irradiation damages and the chemical compatibility with the fuel salt.

II. MECHANICAL PROPERTIES

The Oak Ridge program on the molten salt reactor experiment led to the development of the Hasteloy N alloy, essentially a Nickel ternary alloy added with 8wt% of Cr and 12wt% of Mo. [6] The composition of the alloy was optimized for corrosion resistance, irradiation resistance and high temperature mechanical properties. Pure nickel has a good compatibility with fluorides but lacks the required high strength at high temperature.

Molybdenum, which also has a good compatibility with fluorides, was therefore added in solid solution to nickel to provide high temperature creep resistance and hardening.

The composition of Cr was tailored on the one side to maintain a good corrosion resistance in gas atmosphere containing oxygen due to the formation of a protective oxide. On the other side, the content of Cr was limited in order to suppress voids formation due to the Cr depletion by dissolution in the molten salt.

Helium produced by neutron capture on the isotope ^{58}Ni dominates the issue of the resistance of the material under irradiation. A modified version of Hastelloy N was designed with an improved irradiation resistance due to a fine dispersion of titanium and niobium carbides. These carbides provide coherent interfaces to the nickel matrix which very efficiently traps He atoms.

One can say that within the temperature range envisioned for molten salt reactors at that time (maximum temperature of 700-720°C), there is a first generation structural material that satisfies requested criteria. However, it was also demonstrated that the maximum temperature allowable for this material is of the order of 750°C. Indeed, beyond this threshold, titanium and niobium carbides are dissolved in the nickel matrix. Due to its evolving microstructure, it would therefore be impossible to preserve the material properties required to address the specificity of molten salt reactors at higher temperature.

Replacing molybdenum by tungsten in such alloys could prove beneficial to reach higher in-service temperature from several standpoints. First of all, tungsten diffusion is roughly ten times slower in nickel than molybdenum diffusion [7]. Therefore, there is correspondingly a better creep resistance expected with a Ni-W solid solution than with a Ni-Mo solid solution. This would help to reach higher in-service temperature. Second, a comparison of the ternary phase diagram of Ni-Mo-Cr with Ni-W-Cr shows that there is only one intermetallic phase with a high Cr content. Close to the solubility limit in the low chromium range, there is no embrittling intermetallic in the Ni-W-Cr system. Instead, there is a phase separation between the solid solution and a pure W α-phase (see Figure 1).

The kinetic of precipitation being slow, it allows higher temperature to be used in thermo-mechanical processing for microstructure control with much less susceptibility for intermetallics formation. This can also be used to engineer the microstructure with tungsten precipitates at grain boundaries. This has been done with success for VHTR materials in the seventies. [8] Having such a microstructure with grain boundaries locked in by tungsten precipitates could be another road to process higher temperature materials for molten salt reactors with outstanding mechanical properties. Indeed, such a microstructure with tungsten precipitates at grain boundaries would be stable up to very high temperature. High temperature grain growth would be reduced as well as grain boundary sliding leading to an increased creep resistance. Solubility limits in this ternary system are not well known but are currently being investigated. Preliminary results show that indeed one can precipitate tungsten in the low Cr range as well in reasonable time allowing for the definition of an industrial material. Therefore, Ni-W-Cr alloys look promising for their use in molten salt at high temperature.

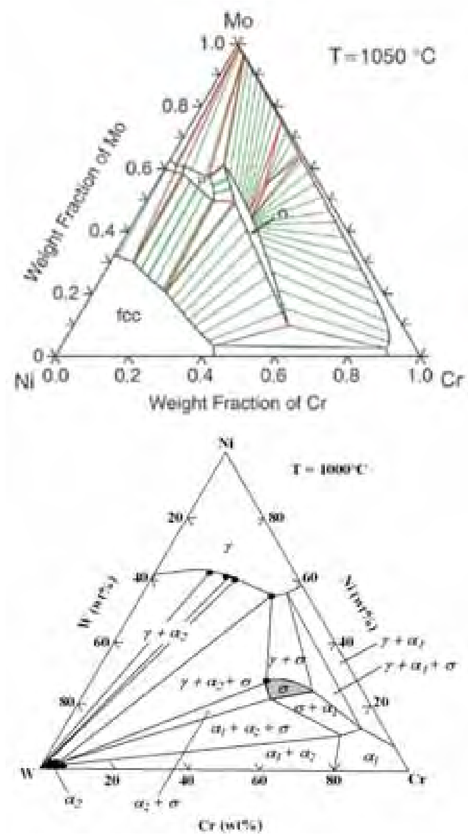

Figure 1: a) Ni-Mo-Cr ternary phase diagram at 1050°C
b) Ni-W-Cr ternary phase diagram at 1000°C

II. NEUTRONIC IRRADIATION DAMAGES

Ni-Mo-Cr alloys were tested under irradiation. The helium formation and diffusion at grain boundaries can be responsible of the metallic alloy embrittlement. The main part of Helium is produced by the action of thermal neutrons on nickel. Niobium and titanium carbides are added to the metallic alloy to trap helium atoms and therefore prevent the alloy embrittlement.

Neutronic calculations have been performed (using MCNP neutron transport code coupled with the lab-made materials evolution code REM) in the case of molten salt reactor operating in a fast spectrum (MSFR) and for Ni-W-Cr alloys which is required for high operation temperature. The neutronic irradiation damages modify the properties of the materials through three effects: the displacements of atoms, the helium production and the transmutation of tungsten to osmium by nuclear reaction. These results obtained for the material damages are presented here for the upper axial reflector [9] which is the most irradiated element in the core, the neutron flux in this reflector being displayed in Figure 2.

III-A- Displacements of atoms

The radiation damages in neutron-irradiated materials depend on many factors (neutron spectrum and flux, irradiation dose) and are expressed in displacements per atom (dpa). That corresponds to the number of times an atom is displaced for a given fluence. The calculations show that the damages are largest in the first two centimeters of the central area (radius 20 cm and thickness 2 cm) of the axial reflector and are quite small, varying from 0.47 dpa/year (for a fuel salt volume of 27 m^3) to 1.17 dpa/year (for a fuel salt volume of 12 m^3).

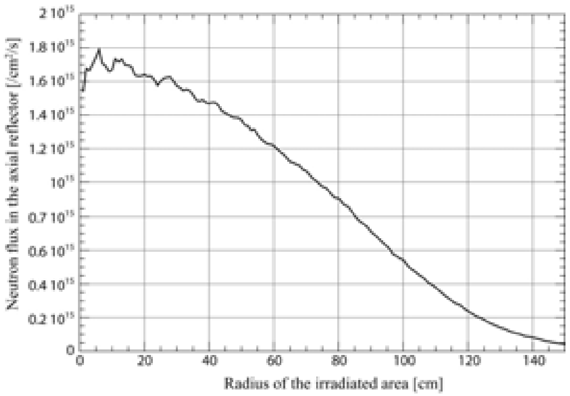

Figure 2: Neutron flux in the axial reflector of the MSFR as a function of the irradiated area considered (zero corresponding to the centre of the core), for a fuel salt volume of 18 m^3

III-B- Helium production

The helium concentration in the structural material is directly determined by its production rates though nuclear reactions. Helium production depends on the boron and nickel amounts in the alloy. It is produced by two nuclear reactions: $^{10}B\,(n,\alpha)\,^{7}Li$ and $^{58}Ni\,(n,\alpha)\,^{55}Fe$. As shown in Figure 3 and as it was previously observed in thermal spectrum (MSRE), the main part of helium is produced by the nickel transmutation.

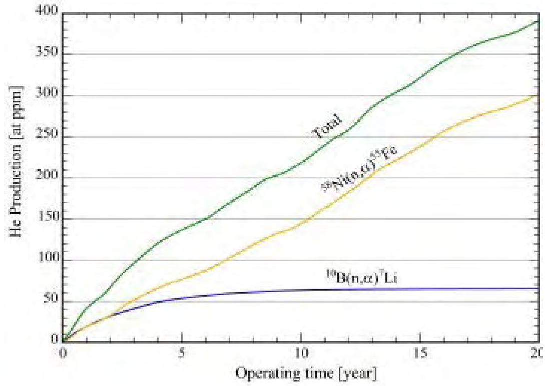

Figure 3: Production of He in the most irradiated part of the axial reflector (radius 20 cm and thickness 2 cm) of MSFR system with a volume core of 18 m^3.

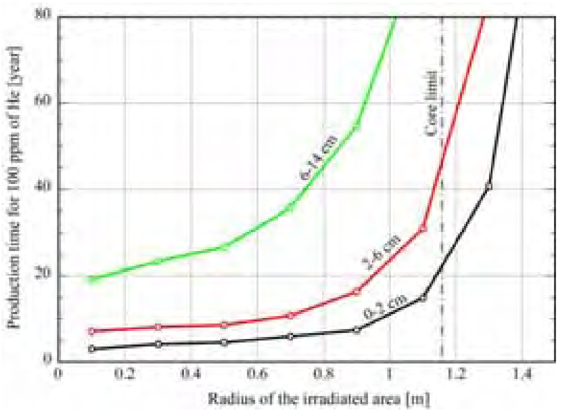

Figure 4: Operation time necessary to produce 100 ppm of He in different depths of the axial reflector as a function of the irradiated area considered (zero corresponding to the centre of the core), for a fuel salt volume of 18 m^3

Moreover, the content of boron can be strongly reduced in the structural material. The largest acceptable amount of helium in the material is not known and the diffusion of helium in Ni-W-Cr alloys has not been yet determined. If we assume that the acceptable limit is equal to a production of 100 ppm of Helium, [15] the equivalent operation time to reach this value are displayed in Figure 4. These operation times, larger than 170 years for the deeper zones of this reflector (from 14 to 30 cm), are not represented. As a conclusion, regular replacements of the most irradiated area of the upper axial reflector have to be planned, but it concerns only its first 15 centimeters.

III-C- Osmium production

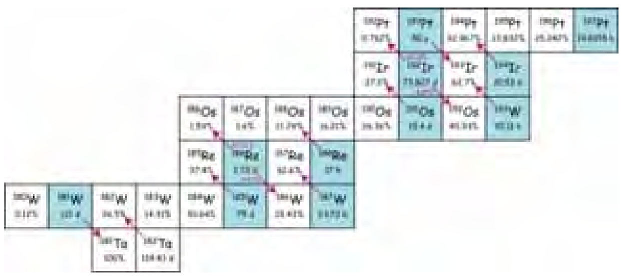

Figure 5: Transmutation cycle of Tungsten, Rhenium and Osmium, due to the neutronic captures; the blue boxes represent the unstable nuclei that decay through the purple arrows

Nuclear reactions lead to tungsten transmutation into rhenium and osmium, as shown in Figure 5. The proportion of transmutation has been calculated and is given in Figure 6 with a neutron flux lower than 1.6 10^{15} neutrons/cm^2/s in the most irradiated part of the structural material (see Figure 2), to be compared to the neutron flux in the core itself which is around 5 times higher.

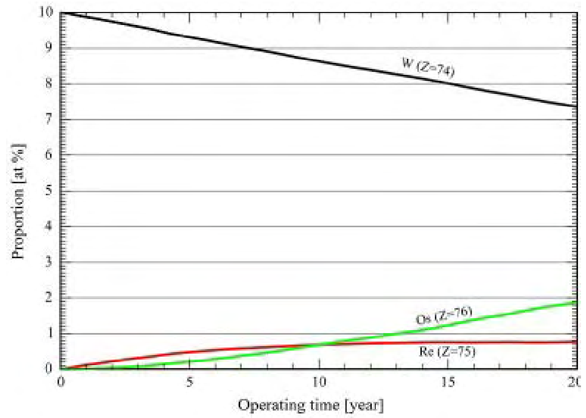

Figure 6: Composition evolution of alloy in W, Os and Rh in the most irradiated part of the axial reflector (central area of radius 20 cm and thickness 2 cm) as a function of operating time, with a fuel salt volume of 18 m^3

Considering that a loss of less than 1at% of tungsten is acceptable, the most irradiated part of the upper reflector has thus to be changed every 5 to 10 years.

Table 1 gathers the results obtained for the different irradiation damages as a function of fuel salt volume, in terms of reactor operation times necessary to produce these damages. The damages are inversely proportional to the fuel salt volume favoring the larger MSFR configurations.

Fuel salt Volume (m^3)	Time (year) (100 dpa)	Time (year) (100 ppm He)	Time (year) (-1at% of W)
12	85	2.2	4.7
18	133	3.2	7.3
27	211	5.5	10.9

Table 1: Reactor operation time necessary to reach a given irradiation damages in the most irradiated part of the axial reflector (central area of radius 20 cm and thickness 2 cm), as a function of the fuel salt volume.

The feasibility of using reflectors made of metallic alloy has been demonstrated from a neutronic point of view. The irradiation damages have been evaluated and the replacement of a part of the axial reflector every five years for example is not a drawback.

IV. CHEMICAL BEHAVIOUR

The chemical behavior of a metallic element or alloy strongly depends on its environment. In the case of fuel salt, the chemical properties of the molten salt will define the chemical behavior of the structural materials. The presence of some fission products can also be responsible of chemical reactions but the main chemical corrosion can be controlled by the control of salt properties.

In Ni-Mo-Cr or Ni-W-Cr alloys, the more easily oxidizable element is Cr because its redox potential is very low. Therefore the main corrosion is due to the dissolution of chromium.

IV-A- Salt properties

The fluoride molten salt is characterized by its redox potential and its oxo-acidity (concentration of free oxide ions in the molten salt). Using thermochemical data, equilibrium diagrams [10] can be calculated giving the stability ranges of different elements in a given molten salt at a given temperature. Such a diagram has been calculated for chromium (Figure 7). This diagram shows the stability of the different chromium-based compounds in their different oxidation states which can be present in the molten salt as a function of the potential and of the oxo-acidity. The oxo-acidity is given by the activity of oxide in the molten salt. For thermochemical calculations, the oxo-acidity is calculated introducing the activity of Li_2O. The relation between the activity of Li_2O and the oxide activity can be determined experimentally and depends on the nature of the molten salt.

The Figure 7 shows that Cr is oxidized for potential higher than -3.5V when the pa(Li_2O) is ranging between 12 and 25. When the acidity decreases (low values of pa(Li_2O) the oxidation of Cr occurs at lower potentials values (between -4.5 and -3.5V).

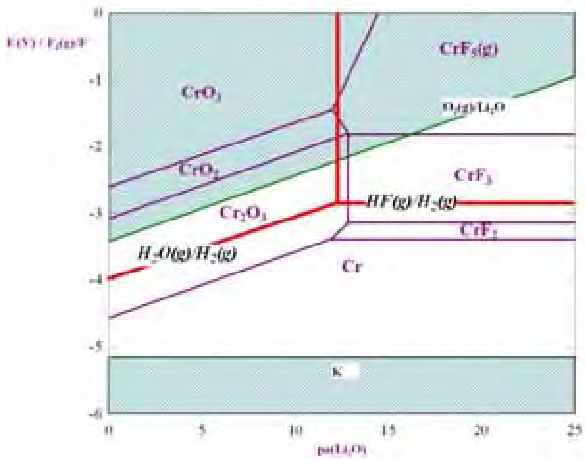

Figure 7: Potential-acidity diagram calculated for Cr in fluoride media at 700°C calculated for activities equal to 1.

Cr can be oxidized to CrF_2 solubilized in the molten salt. Depending on the oxo-acidity of the melt, Cr can also be oxidized to Cr_2O_3. If the oxo-acidity (given in pa(Li_2O) = - log [a(Li_2O)]) of the melt is lower than 13, the oxidation of Cr leads to the formation of the oxide Cr_2O_3 which is known to be protective against corrosion under oxygen atmosphere. However, the high purification of fluoride molten salt contributes to reach very low amounts of oxide (high values of pa(Li_2O)) and in these conditions the chromium oxide is not stable and its oxidation leads to the formation of soluble compounds such as CrF_2.

On the other hand, the high purification of melts against oxide ions is required to prevent the precipitation of solid oxides (such as UO_2 or ThO_2) in the fuel salt. To overcome this dilemma, addition of ZrF_4 was recommended to control the oxide concentration (by precipitation of ZrO_2 insoluble in fluoride molten salt) and the MSRE fuel salt was constituted of 5mol% of ZrF_4.

IV-B- Potential control

The control of redox potential of the fuel salt is the best way to prevent the oxidation of chromium, It is possible to control the redox potential by using a redox "buffer" constituted by the two oxidation states of uranium, UF_4 and UF_3. The fuel potential is given by the following Nernst relation:

$$E\ (V) = E° + (2.3RT/F)\ \log\ ([UF_4]/[UF_3]) \qquad (1)$$

where E° is the standard potential of the redox system (V), R the ideal gas constant (J/mol/K), T the temperature (K), F the Faraday constant (96500C), $[UF_4]$ and $[UF_3]$ respectively the concentrations of UF_4 and UF_3 in the fuel salt.

This ratio can vary from 10 to 100. The lowest limit is given by the solubility of UF_3 and the largest limit by the higher potential acceptable for chromium corrosion.

The dissolution of chromium can be described by the following reaction:

$$Cr + 2UF_4 = CrF_2 + 2UF_3 \qquad (2)$$

The variation of the constant K, which characterizes this equilibrium, with the temperature was experimentally established by Baes [11]. Considering the activity of chromium in the alloy (equal to 0.083 [12]), the concentration of CrF_2 can be calculated for various ratios [UF4]/$[UF_3]$ as a function of temperature (Figure 8).

Figure 8 shows a large difference of the concentration of CrF_2 as a function of temperature. That explains the mass transfer observed experimentally by the ORNL [14] in the convective loops between the hot and the cold parts of the loop. A dissolution of chromium is observed in the hot part of the loop and deposits of metallic chromium are observed in the cold part of the loop. The redox potential applied by the ratio of UF_4/UF_3 is not sufficiently low to prevent the oxidation of metallic chromium in the temperature range of the fuel.

Nevertheless, experimental corrosion tests in molten salt loops have shown that the deposits of chromium were very homogeneous without any dendrites and no clogged pipes were observed.

IV-C- Evolution of fuel potential with operation time

It was demonstrated that a redox potential control of the fuel salt is necessary to limit the corrosion of structural materials. However the potential of the fuel salt increases with the operation time due to the fission reaction. The fissile element in the liquid fuel is uranium. As it was previously described, uranium is dissolved in the fuel under two chemical states, UF_4 and UF_3. When the fission reaction occurs, the fission products are essentially lanthanides (LnF_3) (at an oxidation state III) and gaseous products or noble metals (M) (at an oxidation state 0). The impact of the fission reactions on the chemistry can be schematized by:

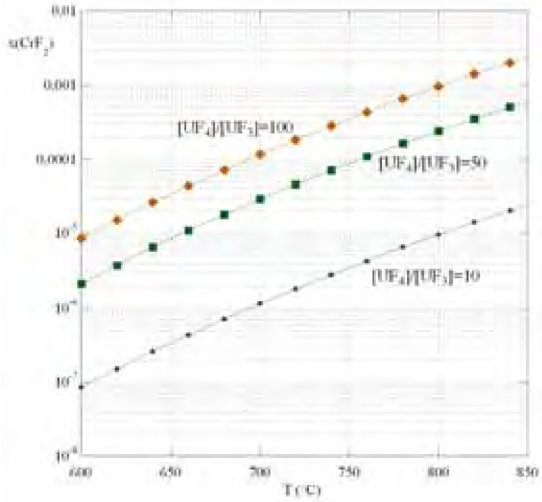

Figure 8: Variation of the concentration of CrF_2 in the fuel salt as a function of temperature for various ratios $[UF_4]/[UF_3]$

$$UF_3 + n \rightarrow LnF_3 + M \qquad (3)$$
$$UF_4 + n \rightarrow LnF_3 + M + 1/2F_2(g) \qquad (4)$$

When the fission occurs on UF_4, it leads to the formation of gaseous fluorine $F_2(g)$. Fluorine gas is an oxidizing element which contributes to increase the redox potential of the fuel salt by the following way:

$$UF_3 + 1/2F_2(g) \rightarrow UF_4 \qquad (5)$$

The consumption of UF_3 as in (5) leads to an increase of the redox potential of the fuel salt according to the relation (1). To decrease the potential during the operation time, a reducing agent is added in the fuel salt: metallic Be in the case of MSRE system and metallic Th in the MSFR concept.

The redox potential has to be well controlled. Indeed, a very low potential value is not desired because in this case, the tritium produced by fission reaction is under gaseous state (TH or T_2) and a large part will diffuse through the structural material in the heat exchangers. When the tritium is under TF state, it is totally extracted by helium bubbling.

IV-D- Corrosion by fission products

Some fission products are corrosive. A small hydrogen production is observed during MSFR operation. The combination of hydrogen with fluorine or oxygen can occur. The chemical form of hydrogen depends both on oxo-acidity and redox potential of the fuel salt. For example, HF and H_2O which can be reduced to H_2(g) by reacting with chromium. Tellurium is also a corrosive product and it is known to be responsible of an intergranular corrosion of Hastelloy N. [13] Tellurium is in the periodic classification in the same column than O or S. Therefore Te, in its metallic state, is an oxidizing agent which can react with metallic chromium. To prevent the oxidation of Cr with Te, it is necessary to control the redox potential of the fuel salt as well as to produce tellurium in its reduced state.

IV-E- Experimental corrosion tests

The large feed-back of MSRE experience and tests in convection loops performed by the ORNL have demonstrated the high resistance against corrosion of Hastelloy N in fuel salt. The corrosion rate is lower than 3µm/year. The Table 2 gives some results obtained in convection loops [14] in LiF-BeF_2-UF_4 or LiF-BeF_2-ThF_4-UF_4 molten salts.

Alloy	**T(°C)**	**Convection**	**Corrosion Rate (µm/year)**
LiF-BeF_2-UF_4			
Hastelloy N modified	676	Natural	0.5
	700		0.9
Hastelloy N Standard	660		1
LiF-BeF_2-ThF_4-UF_4			
Hastelloy N modified	700		0.4
	704-566	Forced 3 to 6 m/s	3
	dT=55		1.5
Hastelloy N Standard	700	Natural	0.5

Table 2: Results of corrosion tests in convection loops obtained by the ORNL

IV. CONCLUSION

A wide range of problems lies ahead in the design of high temperature materials for molten salt reactors. The Ni-W-Cr system looks promising. First results show that such materials have the required properties, especially in terms of compatibility with molten salts and mechanical properties. Their metallurgy and in-service properties need to be investigated in further details regarding irradiation resistance and industrialization.

These damages are dominated by helium production due to the (n, α) reaction on ^{58}Ni. One solution may consist in regularly changing only the first 15 centimeters of the reflectors.

The irradiation damages are logically inversely proportional to the fuel salt volume of the reactor, the smallest volumes (lower than around 15 m^3) only being really disfavored due to a high Helium production.

The chemical corrosion can be controlled by a redox buffer which controls the potential of fuel salt. The redox buffer considered is the redox system UF_4/UF_3. The potential has to be measured on line in the reactor core because the potential increases with operation time due to the fission reaction. Addition of a reducing agent leads to a decrease of the fuel salt potential. The use of an acido-basic buffer to control also the oxo-acidity of the molten salt could stabilize the chromium oxide in the alloy and contribute to the formation of a protecting layer at the alloy surface. The experimental feedback from the ORNL has demonstrated the high corrosion resistance of Ni-based alloys in fluoride molten salts.

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References

1. Merle-Lucotte, E., D. Heuer, *et al.*, Proc. of the International Conference on the Physics of Reactors PHYSOR 2008, Interlaken, Switzermand, (2008)
2. Delpech, S., E. Merle-Lucotte, *et al*., *J. of Fluorine Chem*., 130, 1, 11, (2009)
3. Haubenreich, P. N., J.R. Engel, Nuclear Appl. & Tech., 8, 118, (1970)
4. Grimes, W.R., "Molten salt reactor chemistry", Nuclear Appl. & Tech., 8, 137, (1970)
5. Bettis, E.S., R.C. Robertson, Nuclear Appl. & Tech., 8, 190, (1970)
6. Mc Coy, H.E, *et al*., Nuclear Appl. & Tech., 8, 156, (1970)
7. Tiearnay, T.C., N.J. Grant, Metallurgical transactions A, 13A, 1827, (1982)
8. Tanaka, R., T.K. Ondo, Nuclear Technology, 66, July (1984) 75
9. Merle-Lucotte, E., D. Heuer, *et al*., Proc. Of Advanced in Nuclear Fuel Management IV (ANFM 2009), USA, (2009)
10. Rouquette-Sanchez, S., G. Picard, *J. of Electroanal. Chem*., 572, 173, (2004)
11. Baes, C.F., Proc. of Nuclear Fuel, Conf. 690801, (1969)
12. Grimes, W.R., ORNL report 3708, 238, (1964)
13. Mc Coy, H.E., ORNL report 4812, 201, (1972)
14. Koger, J.W., ORNL report 4832, 124, (1972)
15. Angeliu, T.M., *et al*., *J. of Nuclear Mat*., 366(2007)223

SESSION II SUMMARY / DISCUSSION

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Session II of the GIF Symposium focused on four of the six GIF systems: the GFR, SCWR, LFR, and the MSR. In all, nine presentations on the four systems were given, where each presented an overview of their system's current designs and status, and on priority research being undertaken in support of these systems. Since all four systems were reviewed in one session, it provided a unique opportunity for the audience to consider common issues between the systems in addition to a better appreciation of the unique design and challenges of each system. This was evident in the moderated discussion session. Overall, this moderated session had engaged discussions on four main topics that arose from the presentations.

From the LFR presentations and subsequent discussions, two different opinions on the coolant composition (lead or lead-bismuth) for the LFR reactor were noted. The LFR System is currently considering lead coolant given that there is known challenges associated with the use of a lead-bismuth (LB) based coolant. As such, the audience sought clarification from Mr. Zrodnikov, Co-Chair of the session, on Russia's current plans associated with the LFR development. It was noted that Russia's current efforts on a LFR system would use LB as the coolant and hopes to have a demonstration reactor in the near future (by 2014). The reactor design would be classified a Generation IV design as it would introduce new levels of safety and new advanced technologies. Russia is not currently participating in the LFR System; a decision is to be made in the near future.

During the presentation, a common underlying challenge in developing any one of the four systems presented was associated with materials and the on-going effort required to address these challenges. Given the world's limited resources, and expertise required to address material concerns, participants in the audience noted that better use of research resources may be possible if the different systems could share their material research instead of each system working in solos. Numerous participants noted the potential benefits that could be obtained if a mechanism that would not require the establishment of new legal agreements could be established.

In addition to the amount of material research being undertaken in each system, most of the systems are looking at multiple reactor designs. Given the variety and number of different designs being considered and the effort required to support these designs, it raised the question of whether there is a need to focus the GIF effort only on a few of these designs, given that not every design seems possible. The methodologies presented in the first session could be used as a means to focus the GIF effort to a few designs. Participants noted that this is not the time to reduce research as legal agreements have only recently been established and many projects have just started. As such, it may be better to consider this in 5-10 years, once more results are known. The audience also noted that utilities will decide in the end. The session ended with the audience highlighting the benefits and positive impacts that GIF has made in the world nuclear community. Members of the audience noted that GIF is directly linked to:

- the revived interest in the nuclear industry and engagement of governments in nuclear energy;
- the creation of a framework in which the international community has come together to undertake collaborative nuclear research that:
  - allows for and supports innovation; and
  - supports nuclear research to develop technology solutions.
- the engagement of university participation; and
- providing the means by which the international community came together and agreed to focus R&D effort on six systems, from the approximate hundred systems originally reviewed by the GIF.

SESSION III

SODIUM-COOLED FAST REACTOR (SFR)
INPRO

Co-Chairs: Jean-Louis Carbonnier and Xu Mi

OVERVIEW OF R&D ACTIVITIES FOR THE DEVELOPMENT OF A GENERATION IV SODIUM-COOLED FAST REACTOR SYSTEM

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I. INTRODUCTION

Sodium-cooled Fast Reactor (SFR) nuclear energy systems are among the six candidate technologies selected in the Generation IV Technology Roadmap for their potential to meet the Generation IV technology goals. The primary missions identified for the SFR are (1) contribution to sustainability, in particular through its capabilities for actinide management, and (2) electricity production.

The main characteristics of the Generation IV SFR that make it especially suitable for the missions identified are:

(1) High potential to operate with a high conversion fast spectrum core with the resulting benefits of increasing the utilization of fuel resources.

(2) Capability of efficient and nearly complete consumption of transuranics as fuel, thus reducing the actinide loadings in the high level waste with benefits in disposal requirements and potentially non-proliferation.

(3) High level of safety obtained with the use of active and passive means that allow accommodation of transients and bounding events.

(4) Enhanced economics achieved with the use of high burn-up fuels, fuel cycle (*e.g.*, disposal) benefits, reduction in power plant capital costs with the use of advanced materials and innovative design options, and lower operating costs achieved with improved operations and maintenance.

The SFR can be arranged in a pool layout or a compact loop layout. Reactor size options under consideration range from small (50 to 300 MWe) modular reactors to larger reactors (up to 1 500 MWe). The two primary fuel recycle technology options are advanced aqueous and pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide preferred for advanced aqueous recycle and mixed metal alloy preferred for pyrometallurgical processing.

Owing to the significant past experience accumulated with sodium cooled reactors in several countries, the deployment of Generation IV SFR prototype systems is targeted for 2020. Enhanced economics with high level of safety is deemed be one of the obstacles to early deployment of SFR. This paper provides an overview of the R&D activities currently conducted within the GIF (Generation IV

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International Forum) on the SFR nuclear energy systems.

II. DEVELOPMENT TARGETS

II.A. Development Targets

The SFR is well suited for the management of high-level waste types. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. Water/steam and carbon-dioxide are considered as working fluids for the power conversion system to achieve high level performance on thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR can be competitive on electricity markets. The SFR fast spectrum also makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than in thermal spectrum reactors with once-through fuel cycles

The goals of the SFR R&D program would be achieved by establishing development targets such as economic competitiveness, efficient utilization of resources, reduction of environmental burden and enhancement of nuclear non-proliferation, while maintaining an excellent level of safety(1). The development targets for the Generation IV SFR are summarized as follows:

(1) Safety assurance

The safety design approach for the SFR places the highest priority on preventing the occurrence and evolution of abnormal conditions based on the concept of Defense in Depth. A safety level equivalent to or better than Generation III light-water reactor cycle systems should be achieved.

Passive safety functions should possibly be added or enhanced, and regarding the reactor, measures should be taken for the prevention of any hypothetical core disruption and exclude energetic sequences due to nuclear excursion, in order to ensure that the impact of such a hypothetical accident is confined within the boundary of the reactor vessel or the containment vessel.

The goal of the implementation of these measures is to render the risk of installing the SFR cycle system sufficiently small compared with other risks already existing in society.

(2) Economic competitiveness

For the commercialization of an SFR system, it is important to achieve a level of economic competitiveness that enables the system installation in accordance with market principles. For this purpose, an important goal should be to ensure enough competitiveness in terms of energy cost (unit cost of power generation) compared with the competing energy sources in the future.

(3) Reduction in environmental burden

With the excellent neutron economy characteristics of the SFR, there is a possibility of achieving further reductions in the exposure dose and risks associated with geological disposal, which are already at safe levels, by utilizing the transuranic (TRU) burning characteristics along with implementation of separation and transmutation methods. To this end, the development is advisable for the separation of nuclear transmutation technologies of long-life nuclides [TRU and LLFP (Long-life fission products)] generated by light-water reactors and fast reactors, that would allow the utilization of the full advantages of the closed fuel cycle of the SFR system.

Efforts should also be made for achieving reductions in the amount of waste generated from the operations and maintenance and the decommissioning of system facilities, and the amount of waste migrating to the environment.

(4) Efficient utilization of resources

The capacity for efficient burning of TRU materials, including degraded plutonium, and the excellent neutron economy are some of the advantages of the SFR, which enable the utilization of nuclear energy as a sustainable energy source over a very long time period of more than 1 000 years. Accordingly, the effective utilization of uranium resources includes the recycling of TRU.

The current outlook is that long-term demand for energy will keep increasing on a global scale, but because there is an element of uncertainty in any projection regarding energy supply and demand, an SFR system should possess the flexibility to adapt to changing energy needs by adjusting its actinide management capability (from net consumption to net generation of fissile material).

(5) Resistance to nuclear proliferation and enhanced physical protection

Among the technical features that contribute to the proliferation resistance of the SFR are the characteristics of the recycling process, which include the presence of minor actinides (MA) and highly radioactive (β, γ) fission products (FP) in the recycled fuel, rather than the separation of plutonium. This results in lowering the chemical purity and the fissile fraction of Pu, and in an increase in the surface dose rate of the recycled product. These features enhance the difficulty of accessing the nuclear materials in the fuel cycle and lower their attractiveness, since separated plutonium does not exist in its pure state in any of the system's processes.

Regarding the organizational aspects, it is necessary to implement nuclear safeguards (IAEA safeguards agreements) and to always maintain an accurate material inventory through the utilization of advanced technologies. An advanced system and facility design that allows for the integration of the safeguards and physical protection systems will ensure the implementation of effective accountancy, monitoring and protection measures. It is also necessary to maintain transparency and openness in terms of information in the relationships with external organizations.

II.B. Design Requirements

Eight goals for the Generation IV nuclear energy systems are defined in the four broad areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. The broad design requirements for the SFR system, shown in Table 1, are established in order to satisfy the development targets corresponding to the Generation IV goals. The design requirements are consistent with the Generation IV goals.

SFR Design Requirements		Generation IV Goals	
Breeding Capability	Breeding ratio: ca. 1.2, System doubling time: ca. 30 years	Sustainability	-1:Resource utilization
TRU Burning	TRU burning under fast reactor multi-recycle and long-term storage of LWR spent fuel (Transmutation of LLFP such as I-129, Tc-99 is desirable)		-2:Waste minimization and management
Radioactive Release	Equivalent or less than present LWR application		
PR&PP	Excludes pure-Pu state throughout system flow	Proliferation Resistance and Physical Protection	-1:Minimize diversion or undeclared production -2:Reactors have passive features that resist sabotage
Safety	Operability, maintainability and reparability	Safety and Reliability	-1:operations will excel in safety and reliability
	Active and passive safety		-2:very low likelihood and degree of reactor core damage
	Core damage frequency less than 10^{-6}/ry, Exclude energetic sequence due to excursion		-3:eliminate the need for offsite emergency response
Electricity Generation Cost	Cost-competitiveness with other means of electricity production and a variety of market conditions, including highly competitive deregulated or reformed markets	Economics	-1:life-cycle cost advantage over other energy sources (Low overnight construction cost, Low production cost) -2: level of financial risk comparable to other energy project
Operation Cycle	ca. 18 months, and more		
Construction Duration	As a goal, large-scale: 42 months, medium-scale modular type: 36 months		

Table 1: Major Broad Design Requirements for SFR System

III. SYSTEM DEFINITION

The three options, shown in Figures 1, 2 and 3 displaying respectively loop-type, pool-type and modular-type systems, are under consideration:

- A large size (600 to 1 500 MWe) loop-type sodium-cooled reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.(2), (3)
- A medium or large size (600 to 1 500 MWe) pool-type system also supported by a fuel cycle. (4)
- A small size (50 to 150 MWe) modular-type sodium-cooled reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.(5)

The design and performance parameters of the three options are illustrated in Table 2.

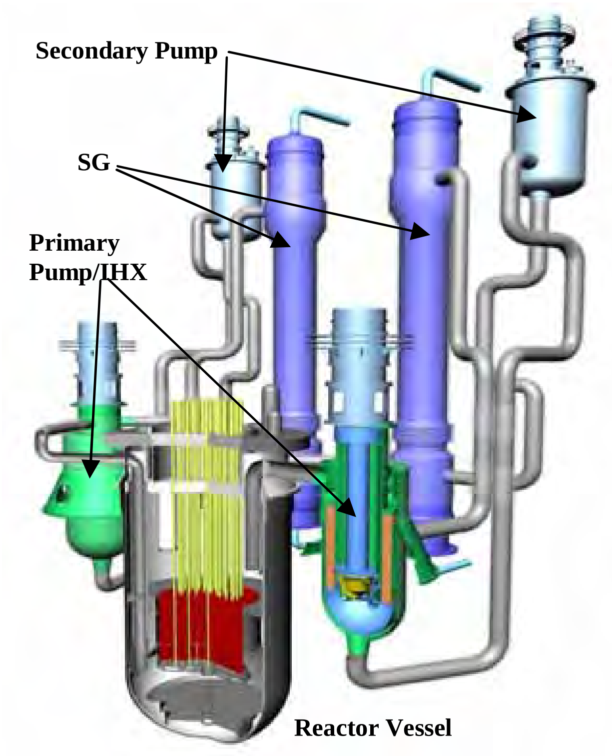

Figure 1: Loop-configuration SFR

SFR Design Parameters	Loop	Pool	Small Modular
Power Rating, MWe	1500	600	50
Thermal Power, MWth	3570	1525	125
Plant Efficiency, %	42	39	~38
Core outlet coolant temperature, ^{o}C	550	545	~510
Core inlet coolant temperature, ^{o}C	395	390	~355
Main steam temperature, ^{o}C	503	495	480
Main steam Pressure, MPa	16.7	16.5	20
Cycle length, years	1.5-2.2	1.5	30
Fuel reload batch, batches	4	4	1
Core Diameter, m	5.1	5.2	1.75
Core Height, m	1.0	0.94	1.0
Fuel Type	MOX (TRU bearing)	Metal (U-TRU-10%Zr Alloy)	Metal (U-TRU-10%Zr Alloy)
Cladding Material	ODS	Mod.HT9M	HT9
Pu enrichment (Pu/HM), %	13.8	14.3	15.0
Burn-up, GWd/t	150	82	~87
Breeding ratio	1.0–1.2	1.0	1.0

Table 2: Design Parameters of Generation IV SFR Concepts

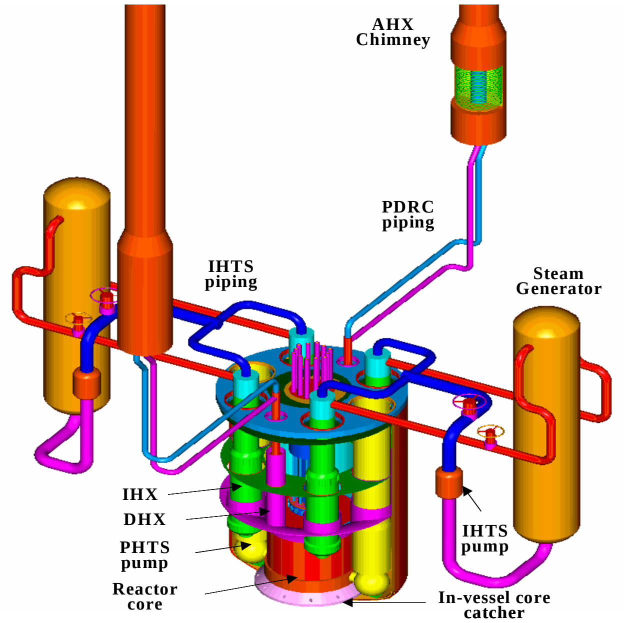

Figure 2: Pool-configuration SFR

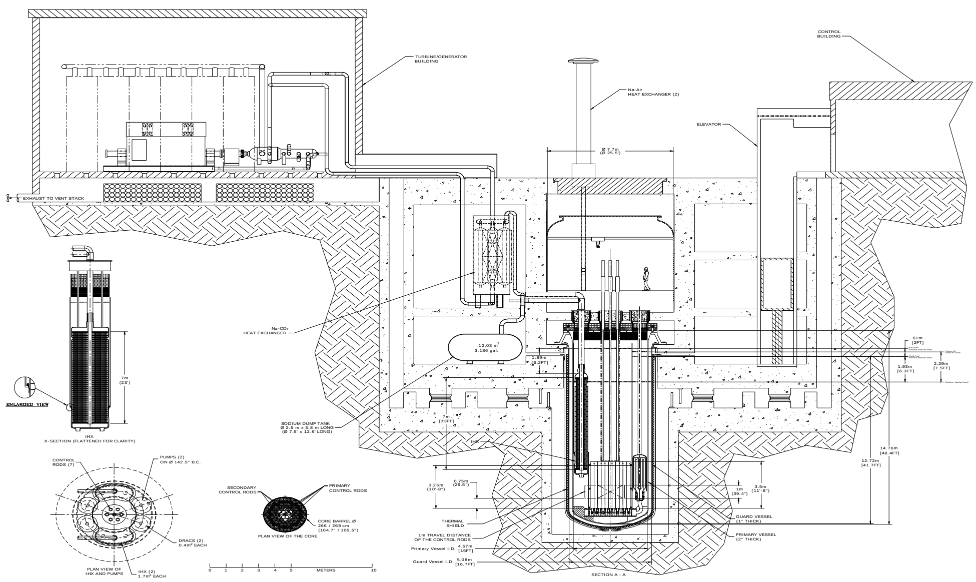

Figure 3: Small modular SFR configuration

IV. R&D ACTIVITIES

IV.A. Status of cooperation

The System Arrangement for the international research and development of the Sodium-cooled Fast Reactor nuclear energy system was signed in November 2006 by Euratom, France, Japan, the Republic of Korea and the United States. In addition, China signed it in March 2009. Three Project Arrangements on Advanced Fuels, Global Actinide Cycle International Demonstration, and Component Design and Balance Of Plant were signed in 2007, and Project Arrangement on Safety and Operation in 2009. Project Arrangement on System Integration and Assessment is expected to be effective in 2009.

IV.B. R&D objectives

The SFR development approach builds on technologies already used in several countries. As a benefit of these previous investments in technology, the majority of the R&D needs for the SFR are related to performance rather than viability of the system. Based on international SFR R&D plans, these research activities have been arranged by the SFR Signatories into five "Projects" to organize the joint GIF research activities:

- System Integration and Assessment (SIA) – The overall objective of this project is to review and integrate the outcomes of the other projects, evaluate their results and assess compliance of the designs under development with GIF goals.
- Safety and Operation (SO) – In order to contribute to the safety assessment of (preliminary) conceptual designs, this project consists of R&D in two areas, namely safety and operation. Experiments and analytical model development are planned in the safety area covering both passive and active safety, and severe accident issues. Options of safety system architectures will also be investigated. The R&D in operation area aims at operation and technology testing campaigns in existing reactors, (*e.g.*, Monju and Phenix) including the end-of-life test in Phenix.
- Advanced Fuels (AF) – This project includes: the development of high-burnup fuel systems (fuel form and cladding) to complete the SFR fuel database; research on remote fuel fabrication techniques for recycle fuels that contain minor actinides and possibly trace fission products.
- Component Design and Balance-Of-Plant (CDBOP) – This project covers the development of the balance of plant for the SFR system. It aims at meeting the GIF criteria in the field of safety, economy, sustainability, and proliferation resistance and physical protection. Experimental and analytical evaluation of advanced in-service inspection and repair technologies including leak-before-break assessment are being carried out. The project includes the development of alternative energy conversion systems with Brayton cycle.
- Global Actinide Cycle International Demonstration (GACID) – This project will demonstrate that the SFR can manage effectively all actinide elements in the fuel cycle, including uranium, plutonium, and minor actinides (neptunium, americium and curium). This technical demonstration will be pursued in a reasonably short time frame using existing fast reactors.

IV.C Milestones

The key dates defined in the five R&D projects of the SFR system are as follows:

- SIA Project
  Definition of SFR System Options

  2009: Initial specification of SFR system options

  Assessment of SFR System Options

  2009-2010: Compile self-assessment results for SFR system options

2009-2010: Compile contributed trade studies proposed by members

Definition of SFR Research and Development Needs

2008: Review and refine SFR R&D needs in the SRP

2009: Review of existing Project Plans to glean R&D needs and gaps

2010: Integrate R&D results to refine the system options & Assess R&D results to provide feedback (guidance) to technical R&D Projects.

- SO Project

R&D for Safety:

2008-2009: Preliminary Assessment of candidate safety provisions and systems

2008-2012: Performance assessment of safety provisions and systems

2011-2015: Qualification of safety provisions and systems

R&D for Reactor Operation and Technology Testing:

2008-2011: Tasks related to SIA Project

 * Phenix end-of-life program
 * Thermal-hydraulics/General system
 * Feedback of the decommissioning of LMFR

2008-2012: Tasks related to CDBOP Project

 * In service inspection technique development from existing reactors to future SFR
 * Sodium chemistry
 * Sodium technology

- AF Project

2006-2007: Preliminary evaluation of advanced fuels

2007-2010: MA fuels evaluation

2011-2015: High-burnup fuel behavior evaluation

2016: Demonstration & Application of advanced Fuel head-end process in the fuel cycle backend

- CDBOP Project

2007: Viability study of proposal concepts

2007-2010: Performance tests for detail design specification

2011-2015: Demonstration of system performance

- GACID Project

2007-2012: Preparation for the limited minor-actinide-bearing fuel preparatory irradiation test

2007-2012: Preparation for the licensing of the pin-scale curium-bearing fuel irradiation test

2007-2012: Program planning of the bundle-scale minor-actinide bearing fuel irradiation demonstration

IV.D Main activities and outcomes

The SFR System Steering Committee (SSC) was formally organized in September 2006 to plan and carry out the research and development work necessary to establish the viability and to optimize the performance of the SFR System, and facilitate the eventual demonstration of the SFR System. Since then, the SFR SSC has developed and revised a comprehensive SFR System Research Plan.

Activities on Integration and Assessment were conducted jointly with the SFR SSC and the provisional SIA Project Management Board (PMB) for the SIA Project aiming at clarifying the project objectives, identifying integration and assessment work to be performed, and defining the relationship between technical PMBs and concept developers. The integration function of this Project will cover a review of the results from technical Projects aiming at their integration, regular updating of the systems options and establishment of a comprehensive list of R&D needs. The Project Plan is expected to be finalized in 2009 in order to complete the Project Arrangement negotiations covering the implementation of the unique aspects of this Project.

Regarding the SO Project, collaboration was launched early in 2009 after the signature of the PA.

Four options are being considered within the AF project for the SFR fuel: oxide, metal, nitride and carbide. Various fuel irradiation tests were ongoing in 2007 aiming at selecting advanced fuel options. Reactors available for those irradiation tests include Phenix in France, ATR in the United States and Joyo in Japan. Fuel evaluation studies and analytical work using fuel performance codes are in progress based on available information from previous tests including fuel property measurements and irradiation tests. The fuel evaluation covers minor-actinide-bearing fuel performance, minor-actinide-bearing fuel fabrication and high burn-up capability. The results will support the selection of advanced fuel options.

Within the CDBOP project, a program of sodium tests with external ultrasonic sensors is being defined in France for the study of *in-situ* inspection and repair technologies. Results of a feasibility study for under-sodium visualization technologies will be reported by the Republic of Korea. A feasibility study of alternative energy conversion system concepts, thermodynamic cycle evaluation coupled with an SFR is being implemented in France. The United States are contributing results of compact heat exchanger test for super-critical CO_2 Brayton cycle, closed Brayton loop test and analysis. Japan is providing results of preliminary design study of plant system adopting supercritical CO_2 turbine system, thermal-hydraulic test, liquid sodium/CO_2 reaction test, and material corrosion test under supercritical CO_2 flow.

The joint activities within the GACID project focused on the evaluation of minor-actinide-bearing fuel material property, and analysis and evaluation of irradiated-fuel data in 2007. In addition, preparation for minor-actinide-bearing fuel material property measurement (high Am content fuel and Cm-bearing fuel) was performed in France. Also, raw material preparation for material property measurement, and preparation for MA-bearing fuel material property measurement (supplemental data) was carried out in the United States. Finally, Japan contributed results from previous irradiation tests in Joyo (*e.g.* Am-1 test) and carried out preparation for minor-actinide-bearing fuel material property measurement (low Am content fuel).

V. CONCLUSION

The international collaborative R&D activities for SFR system within GIF are being successfully conducted aiming at the deployment targeted for 2020.

References

1. Ichimiya, M., *et al*., “Overview of an R&D Plan for the Development of a Generation IV Sodium-Cooled Fast Reactor System,” Proceedings of ICAPP05, Seoul, Korea, May 2005.

2. Aoto, K., H. Kamide, H. Oshima, I. Sato, H. Hayafune, H. Niwa and M. Morishita “Current Status of Japanese Sodium Cooled Loop Type Fast Reactor (JSFR) Development,” Proc.ENC2007, Brussels, Belgium (2007).

3. Mizuno, T., *et al*., “Advanced Oxide Fuel Core Design Study for SFR in the Feasibility Study in Japan”, Proceedings of Global 2005, Tsukuba, Japan, 9-13 October 2005.

4. Hahn, D., *et al*.," Conceptual Design of the Sodium –Cooled Fast Reactor KALIMER-600", Nuclear Engineering and Technology, Vol. 39, No. 6, pp. 205-218(2007).

5. Chang, Y., M. Konomura, P. Lo Pinto, "A Case for Small Modular Fast Reactor", *J. of Nuclear Science and Technology*, Vol. 44, No. 3, pp. 264-269(2007).

INTERNATIONAL PROJECT HARMONIZATION FOR SFR DEVELOPMENT

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I. INTRODUCTION

In January 2008, the U.S Department of Energy (DOE), the French Commissariat a l'Energie Atomique (CEA) and Japan Atomic Energy Agency (JAEA) expanded cooperation on Sodium-cooled Fast Reactor (SFR) prototype development through a Memorandum of Understanding (MOU) signed by former DOE Assistant Secretary for Nuclear Energy Dennis R. Spurgeon, former CEA Chairman Alain Bugat and JAEA President Toshio Okazaki. [1] The MOU established a collaborative framework for the three research agencies (hereinafter the "participants") to jointly cooperate with the ultimate goal of deploying sodium-cooled fast reactor prototypes.

In signing the MOU, each of the parties affirmed its intent to develop advanced fast reactor prototypes according to its respective national program's objectives, and recognized that each country's individual development of SFR technology should not be duplicative. The participants entered into the MOU because of their common interest in developing SFRs in roughly the same timeframe and the recognition that technical expertise, resources and infrastructure required to deploy sodium-cooled fast reactor prototypes could be shared in a mutually beneficial manner.

This paper summarizes the progress made under the MOU and outlines one approach to effectively supporting infrastructure activities needed to deploy initial SFR prototypes and coordinating future technology development with the long-range research and development collaboration being performed under the Generation IV International Forum (GIF). It aims also to do so in a complementary fashion to facilitate the subsequent commercialization of SFR technology.

Recently, the U.S. fuel cycle research and development program has shifted from a near-term technology deployment program to a long-term, science-based research program. As a result, the U.S. is not currently pursuing the development of a commercial SFR prototype within the next two decades. [2]

II. BACKGROUND

The U.S., France and Japan also cooperate under the GIF which furthers the research and development of future nuclear energy systems. The United States first proposed the Generation IV concept in 1999 and the Generation IV International Forum (GIF) was created when Argentina, UK, Canada, Korea, Japan, Brazil, France and South Africa signed the GIF charter in July 2001. Since then, Switzerland, EURATOM, China and Russia have also signed the GIF charter.

In this framework, six next generation reactor types were selected in July 2002, which include the Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), Sodium-cooled Fast Reactor (SFR), Super Critical Water Reactor (SCWR) and Very High Temperature Reactor (VHTR). The progression of R&D activities for these reactor designs is divided into three phases. The first is the *viability* phase, where the principal objective is to resolve key feasibility and proof-of-principle issues. The second phase is the *performance* phase, where the key subsystems (such as the reactor, recycling facilities or energy conversion technology) need to be developed and optimized. The third phase is the *demonstration* phase, which has a number of options depending on the nature of the participation of industry, government, and even other countries in the project. The scope of Generation IV R&D is focused on the viability and performance phases. [3]

In the case of the SFR, EURATOM, France, Korea, United States and Japan signed the SFR system arrangement in 2006. Russia and China joined as observers. In March 2009 China signed the system arrangement and is now a participating country.

In 2006, major steps towards SFR development were taken in three of the participating countries as shown in Figure 1. In January 2006, the French president announced a national project which includes a fourth generation prototype reactor operation in 2020; SFR is thought to be a strong option for this prototype reactor. [4]

In February 2006, the United States proposed the Global Nuclear Energy Partnership (GNEP). GNEP has grown to an international framework with 25 partner nations in pursuing the expansion of clean, sustainable, nuclear energy worldwide in a safe and secure manner, while at the same time reducing the risk of nuclear proliferation. [5] The U.S., France and Japan also cooperate within the framework of GNEP. As part of the domestic GNEP program, the U.S. pursued the SFR for near-term deployment as part of a closed fuel cycle.

In Japan, “Feasibility Study on Commercialized Fast Reactor Fuel Cycle Systems” was conducted from 1999 to 2006. Based on the feasibility study, “Fast Reactor Cycle Technology Development Project (FaCT)” which targets a demonstration SFR plant construction in 2025 has been activated since April 2006. [6]

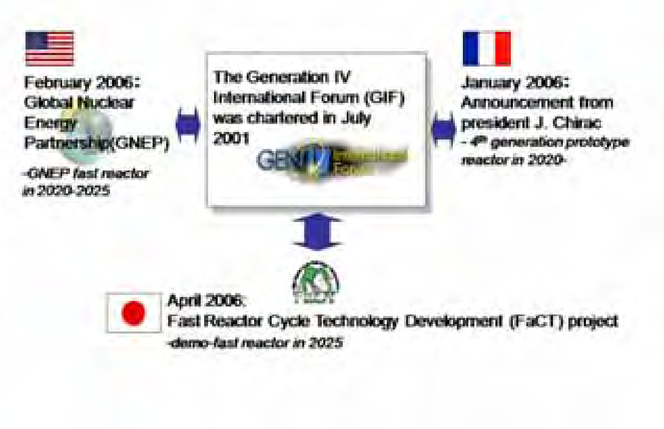

Figure 1: Outline of SFR Development Circumstances

III. SUMMARY OF ACTIVITIES UNDER THE MOU

Section III provides a summary of cooperation progress [7] achieved by the participants in the first year of the MOU.

III.A. Overview of the Memorandum of Understanding

As stated in the signed MOU, each participating country was committed, as part of their national programs, to developing SFRs to advance their respective countries’ intention of building demonstration/prototype (hereinafter called “prototype”) reactors within the next two decades toward the ultimate goal of commercial deployment. Therefore, the MOU initially focused on coming to a common understanding of the mission and requirements for an SFR, on various fuel types for a fast reactor system, and on how each country’s infrastructure, whether existing or proposed, could support fast reactor technology development.

Under the MOU, the participants shared the intention to outline a collaborative framework, review the reactor design criteria, and hold workshops and discussions to reach common recognition on reactor requirements, toward the ultimate goal of deploying SFR prototypes through an efficient collaborative process.

In addition, the participants explored options for leveraging the use of existing, new or refurbished support facilities for component testing, fuel development, and safety testing.

The work conducted under the MOU directly addressed one of the GNEP objectives: "To develop, demonstrate, and in due course deploy advanced fast reactors that consume transuranic elements from recycled spent fuel." [5] Repeated recycle in fast reactors was considered necessary to meet the overall GNEP waste management and proliferation objectives. Furthermore, fast reactor recycle would extend uranium resources.

The work activities under the MOU were organized into seven tasks. Task leads were designated from each participant to conduct the work activities associated with each task. The following shows the scope of each task.

(1) Establishing design goals and high level requirements for the prototypes.

(2) Defining common safety principles.

(3) Discussing the power level and configuration of sodium-cooled (loop and pool) fast reactor.

(4) Preliminarily comparing oxide and metal fuels and assessing the advantages and disadvantages of each.

(5) Discussing a common strategy about fuel facilities needed to provide start-up fuel to the prototypes.

(6) Identifying key technical innovations to reduce capital, operating and maintenance costs.

(7) Identifying test and support facilities and establishing a plan for securing the infrastructure needed to support materials, components and safety testing for the prototypes.

In addition, the participants exchanged information on their national programs in order to begin to develop target dates for prototypes to be used for planning purposes. This addresses one of the areas of cooperation from the MOU: "discussing a draft schedule of target dates for prototypes, including possible initial reactor start-up and full power operations to use as a planning basis; this schedule should be consistent with the national programs of the participants' countries."

III.B. Design Goals, Safety Principles and High Level Requirements (Task I & II)

The participants developed mission objectives for a generic concept, which was called the AFR (Advanced Fast Reactor). The AFR has the following five mission objectives:

(1) Demonstrate TRU recycling while generating electricity, thereby demonstrating sustainable electricity generation.

(2) Demonstrate fast reactor safety.

(3) Demonstrate design features for cost reduction and financial risk minimization.

(4) Provide capability for fast spectrum irradiations.

(5) Demonstrate reactor safeguards and security.

The primary mission of the first AFR prototype is to demonstrate the waste management and resource utilization benefits through the repeated recycle of transuranics, while generating electricity.

The transmutation of TRU is accomplished by fissioning and this is most effectively done in a fast neutron spectrum. Therefore, the AFR will be a fast-spectrum reactor. Sodium is the most proven coolant for fast reactors and was selected as the coolant. Multiple prototypes may be required to fulfill all mission objectives and support commercialization.

Task I and II were combined and entailed discussions among U.S., Japanese, and French experts leading to a draft document that provides high level requirements for a SFR prototype, together with the top level safety design principles and objectives. The Electric Power Research Institute (EPRI) document: *Advanced Light Water Utility Requirements Document* was used as a starting point for the design goals discussions. Goals or requirements that are specific to one country were identified and highlighted. In addition, the document: *Requirements for a Standard Commercial Advanced Burner Reactor* generated by the U.S. was also considered. To the extent practical, differences between the requirements for the prototype and future commercial plant were identified. In the area of safety design, discussions focused on defining a set of common safety principles to guide the design selection process, including identification of key safety design goals and quantification of reactor/plant safety performance requirements.

III.C. Power Level and Reactor Configuration Studies (Task III)

Technical specialists in Japan, France, and the U.S. compared pool and loop configurations of fast reactor plants. Discussions focused on understanding the characteristics of pool and loop SFR plants and generating a list of the similarities and differences of these two plant configurations and also understanding the innovations that can be introduced into the plants to improve the pool and loop concepts.

A criteria matrix was developed and discussions were held to facilitate comparisons and reach consensus. The criteria matrix was completed for innovative pool and loop plants to understand the improvements that can be made to these systems to improve safety, reliability, and economy.

Because the U.S. and France do not have a specific design for the AFR like the Japanese JSFR, the U.S. and France started with the high level requirements generated in Task I & II (discussed in Section III.B) and then generated a list of the main systems and components that fulfill those requirements. The high level requirements were grouped into three areas as agreed to by the participants:

- Safety and Investment Protection
- Reliability, Operability, and Maintainability (Fabrication and Construction, Inspection and Repair)
- Economics and Availability

For each requirement, the design features that contribute to fulfilling that requirement were listed and compared.

Regarding the power level of SFR prototypes, the participants are evaluating initial plant ratings ranging from about 100 to 750 MW electric.

III.D. Sodium-cooled Fast Reactor Fuel Type Comparison (Task IV)

This task provided an assessment of advanced SFR fuels needed for start-up fuel and transmutation (or minor actinide (MA) bearing) fuels. The comparison assessed the current state of understanding of the primary fuel options as well as an assessment of the fabricability, steady-state performance, off-normal performance, and the ability to recycle potential TRU fuel forms. It was not the intent of this effort to make a selection of a particular fuel form but to provide the needed basis and data for a fair comparison of fuel types and associated fuel cycles or identify the areas where data is lacking.

This task included two primary efforts. First, because the fuel cycle strategy differs between each participating nation, a description of the current fuel cycle strategy was provided from the perspective of fuel selection including a summary of the major features of the concept prototype SFR fuel and core design. Second, the participants identified areas to be evaluated during the fuel selection process, including performance, high burnup capability, licensing criteria, fabrication, and recyclability.

III.E. Start-up Fuel Fabrication Requirements and Facility Options (Task V)

The national strategies of the United States, Japan, and France were identified and the possible strategies for obtaining start-up fuel for a prototype SFR in each of the three nations were described. This activity identified possible areas of cooperation and harmonization to achieve the most cost effective strategy. At that time, the three countries had similar goals, development schedules, and deployment time-tables including:

- Fabrication capacity
- Start-up fuel without minor actinides (*i.e.*, using conventional fuels)
- Schedules for 2017-2030
- A lack of existing facilities to fully address the needs
- Mixed oxide fuel is a strong option for the participants

As noted earlier, the current U.S. program has shifted its focus and timetable from near-term fast reactor deployment to long-term fuel cycle research.

III.F. Technology Innovations for SFR Cost Reduction (Task VI)

Cost reduction for SFR technology is an important goal in each participant's domestic reactor technology development program that supports a long-term commercialization of the technology. Research and development activities would include:

- innovative technologies to reduce the capital cost of the reactor plant systems and
- innovative features to improve the reliability, maintainability, and longevity of the reactor plant systems that impact on operating costs.

This task identified the predominant cost reduction technologies being pursued by each country's program and potential research and development activities of common interest.

This work was performed in a three-step process. First, a list of innovative features or technologies being developed by each party was produced; this included an indication of the relative priority and near-term funding plans for research and development. Second, these documents were exchanged, specific technologies of common interest were identified, and a consensual evaluation was conducted of the relative promise and development time of each innovation. The technology list was prioritized based on both demonstration timing (near-term) and maximum benefit (long-term). Third, ideas for collaborative projects and/or exchange of key development data were recommended.

III.G. Infrastructure Collaboration (Task VII)

This task followed a strategic plan that consists of the following steps:

(1) Identify the research and development that is needed to develop SFR prototypes.

(2) Identify existing infrastructure that can be used to conduct the needed research and development.

(3) Identify the gaps between the needed research and development and the capabilities of the existing infrastructure.

(4) Define infrastructure projects that could be used to bridge the gaps.

(5) Decide and agree on implementing these projects.

Infrastructure projects have been identified to fill gaps (Step 4). These include a critical facility, an experimental reactor for transient testing, and sodium loops for component testing. The decisions on proceeding with specific infrastructure collaboration projects are being considered and the necessary implementation agreements will be developed in the future (Step 5). Infrastructure collaboration will be an ongoing activity. Each step in the strategic plan will be revisited as work proceeds and the participants evaluate their progress in the development of SFR prototypes.

IV. SFR HARMONIZATION

Two aspects of international SFR cooperation are shown in Figure 2. For SFR development, since most of the experimental and prototype fast reactors have already been shutdown, international collaboration using residual resources is very important. In Japan, there exist Joyo and Monju. Their availability for fast neutron irradiation is now getting more and more important because other reactors like French Phenix and US EBR-II have been shutdown. The US has the Transient Reactor Test facility (TREAT), currently in shutdown, for potential future transient irradiation test and Japan is strongly interested in it.

Figure 2: Overview of SFR International Cooperation

In the case of sodium component development and near term fuel development, most of them are based on rather conventional technologies even if new designs include some advanced parts and the component scales are larger than that of conventional ones. The historical experience gained by the US, France and Japan in their SFR development programs is extremely important and provides a mature technology basis for future collaboration. From the view point of collaborative utilization of R&D facilities, US, France and Japan have major facilities which could be utilized for large-scale sodium component development and fuel development tests (*e.g.* irradiation tests, transient overpower tests). Considering the above reasons, the participants chose to collaborate in accordance with the MOU described earlier.

The participants are also engaged as part of the GIF initiative. The GIF provides wider international collaborative framework involving several countries. This framework is thought to be suitable for broad and long-term R&D items like advanced fuel development and advanced energy conversion systems. In particular, the SFR System Arrangement has the objective to plan and carry out the research and development work necessary to establish the viability and to optimize the performance of the SFR System, and to facilitate (but not to undertake) the eventual demonstration of the SFR System. The Generation IV goals are shown as follows:

<Sustainability>

(1) Generate energy sustainably, and promote the long-term availability of nuclear fuel

(2) Minimize nuclear waste and reduce the long term stewardship burden

<Safety & Reliability>

(3) Excel in safety and reliability

(4) Have a very low likelihood and degree of reactor core damage

(5) Eliminate the need for offsite emergency response

<Economics>

(6) Have a life cycle cost advantage over other energy sources

(7) Have a level of financial risk comparable to other energy projects

<Proliferation Resistance & Physical Protection>

(8) Be a very unattractive route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism

The Generation IV scope includes a wide range of systems with various coolants (gas, sodium, lead, super critical water) and fuel types (oxide, metal, carbide, molten salt). In the case of the Generation IV SFR, the scope includes advanced fuels and advanced energy conversion systems like carbide fuel and super critical carbon dioxide Brayton cycle. These advanced technologies have not accumulated enough experience in the past or existing SFRs to demonstrate a sufficient level of technical maturity and are more relevant to the feasibility or performance phases. Therefore these and similar advanced technologies are considered to be suitable to the Generation IV framework and collaborative development of those advanced technologies will contribute to strengthening SFR potential as a future generation system.

It is prudent for the three countries to explore various approaches to continue effective collaboration, consistent with each of their national programs. Figure 3 depicts one model for future cooperation that includes evolutionary technology development activities, based on relatively mature technologies focused on cost reduction along with mutually beneficial infrastructure development and use. This approach would be particularly useful in supporting the deployment of demonstration facilities with the potential to accelerate basic SFR technology development as well as contributing to and informing the longer-term research efforts under the GIF. A new intergovernmental agreement among the three countries, building upon the MOU, would be needed to fully implement this approach.

The longer-term and broader research and development focused on viability and performance is more suitable to the GIF framework. The U.S., Japan, and France will continue to support research under the GIF SFR system arrangement. The System Arrangement recognizes that research in this area will be pursued on a bilateral or multilateral basis.

In particular, the U.S. will continue to be fully engaged in the GIF SFR activities, along with other bilateral or multilateral international arrangements, pending any future decisions to resume the pursuit of a domestic prototype fast reactor in the U.S.

France and Japan, will, on their side, also continue to be fully engaged in the GIF SFR activities, all while pursuing their national programs to construct prototype or demonstration fast reactors, which includes investigating possibilities for multi–lateral cooperation.

Although the future deployment timetable for a U.S. fast reactor is uncertain, the participants recognizing the importance and effectiveness of the trilateral framework, intend to analyze the national plans and milestones, along with the results from the seven tasks conducted under the trilateral MOU, and consider various options to enhance complemen–tarities among them in subsequent collaborations.

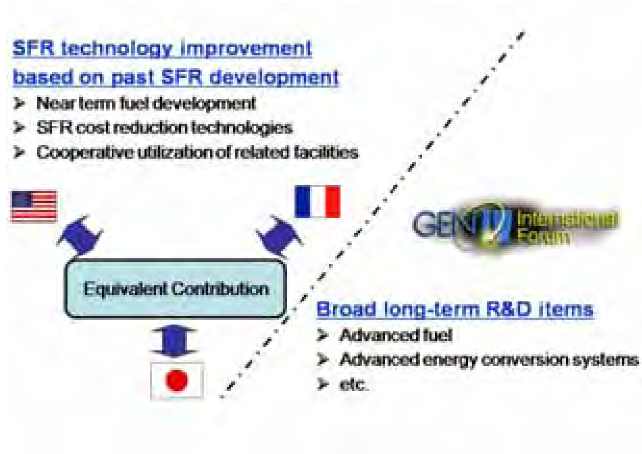

Figure 3: Harmonization of International Projects

V. CONCLUSION

The DOE, CEA and JAEA which participate in the GIF SFR activities have been cooperating on the development of SFR prototypes through the MOU since January 2008. The MOU initially focuses on reaching to a common understanding of the mission and requirements for an SFR, comparing various fuel types of a fast reactor system, and assessing each country’s infrastructure, either existing or proposed, that could support fast reactor development. Although the United States has recently suspended its plans for the prototype fast reactor construction within the next two decades, such collaboration would be useful in supporting demonstration facilities with the potential to accelerate basic SFR technology development. It would also contribute to enhancing a long-term SFR research and development under the GIF framework involving a broader array of countries.

References

1. Press release from USDOE on February 1, 2008, United States, France and Japan Increase Cooperation on Sodium-Cooled Fast Reactor Prototypes.

2. U.S. Department of Energy FY2010 Congressional Budget Request, ref. p. 625.

3. Generation IV International Forum, 2008 Annual Report.

4. World Nuclear Association (WNA), Nuclear Power in France, *www.world-nuclear.org/info/inf40.html/*.

5. USDOE, Global Nuclear Energy Partnership (GNEP) Steering Group Action Plan; An Action Plan for the Safe, Secure Global Expansion of Nuclear Energy, Adopted 13, December 2007.

6. Sagayama, Y., Launch of Fast Reactor Cycle Technology Development Project in Japan, (GLOBAL 2007), Boise Idaho, September 9-13, 2007.

7. JAEA, DOE and CEA, Summary of Progress Concerning Cooperation on Sodium-Cooled Fast Reactor Prototypes, December (2008).

SFR STATUS FOR ONGOING RESEARCH AND RESULTS: ADVANCED FUELS

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I. INTRODUCTION

The Advanced Fuel project in the framework of the SFR system development aims at investigating high burn-up minor actinide bearing fuels as well as claddings and wrappers withstanding high neutron doses (>200 dpa) and temperatures (core outlet temperatures: 550°C). The R&D topics of the Advanced Fuel project deal with fuel fabrication, fuel behavior under irradiation as well as pin clad and wrapper materials developments. (The feasibility demonstration of Uranium, Plutonium and Minor Actinides in the fuel cycle is addressed within the frame of a different project: Global Actinide Cycle International Demonstration (GACID) [1])

The Advanced Fuel project started on 21 March, 2007. It is conducted by CEA (France), DOE-INL (USA), JRC-TUI (Europe), JAEA (Japan) and KAERI (Korea).

II. CONTENT AND SCHEDULE OF THE AF PROJECT [2]

The main challenge for fuel developments for future SFR systems is the development and qualification of a nuclear fuel element (fuel, clad and wrapper types, compositions and designs) which meets the GIF goals. That means: achieving high burn-up (~20 at %), operating at high temperatures and recycling minor actinides into the fuel. High burn-ups will allow uninterrupted reactor operation over long periods of time and consequently, reduce spent fuel volumes, operation costs and eventually fuel cycle costs. High burn-ups are however associated with physical limitations, including dimensional stability of core materials, Fuel-Cladding Mechanical Interactions (FCMI) and/or Fuel-Cladding Chemical Interactions (FCCI), due to the swelling of the fuel. High temperatures will enhance the energy conversion ratio and consequently, the economic competitiveness of the reactor. High temperatures lead to further challenges for fuels and core materials developments too. Minor actinide incorporation aims at reducing the actinide content in the high level waste and consequently brings benefits in disposal requirements and potentially non-proliferation. Since americium is a strong gamma emitter, and curium a high neutron emitter, minor actinide incorporation in the fuels will necessitate shielding, remote operations by robots and simplification of the fuel fabrication process. Moreover the high volatility of Am components has to be managed during fuel fabrication and irradiation phases, where Am should be more

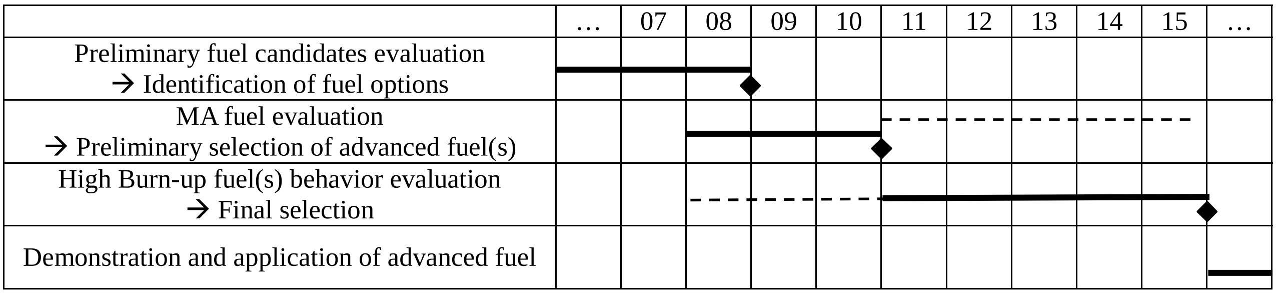

Figure 1: Main phases and milestones of the SFR Advanced Fuel project.

readily redistributed within the fuel than other actinides. Finally, the significant Helium production during fuel irradiation (related to ^{241}Am transformations) which can involve fuel swelling, degradation of the thermal properties and high pressurization of the pins is another major issue to be managed.

Based on background knowledge and past SFR experience, oxide, metal as well as nitride fuels (and more recently carbide[1]) are candidates under consideration. Oxide Dispersion Strengthened (ODS) Ferritic and Ferritic/Martensitic steels are the reference materials for the cladding and the sub-assembly wrapper.

The project consists of 3 major steps until 2015:

- The first phase deals with a preliminary evaluation of the candidate options (2007-2008) in order to identify the capability and the applicability of the fuels and the materials with respect to minor actinide incorporation therein, high temperature operation and high burn-up irradiation behavior. The evaluation concerns fabrication processes and behavior under irradiation. It aims at determining the next major steps to be undertaken for the use of these fuels too.

- After the identification of the advanced fuel options, major R&D efforts will be focused on fabrication feasibility and irradiation behavior of Minor Actinides bearing fuels (2008-2010). A preliminary selection of advanced fuel(s) will then be made.

- The third phase (2011-2015) will consist of the assessment of the high burn-up capability of advanced fuel(s) and materials.

The culmination of this path of research and development will lead to the demonstration and application of the advanced fuel design(s) in the SFR.

The steps and the schedule of the Advanced Fuel project are summarized in Figure 1.

Remark: Fuels under consideration up to now have been mixed Uranium-Plutonium based fuels as SFR driver fuel with MA incorporation up to a few percent, in accordance with the so-called homogeneous MA recycling in nuclear systems. The heterogeneous route for MA transmutation, for which MA will be concentrated in the fuel of the radial blankets is now under discussion with the System Integration & Assessment project and could be included within the frame of the AF project in the future.

III. WORK PERFORMED AND RESULTS

Information from background knowledge, past SFR experience as well as ongoing national and collaborative SFR programs regarding cladding and wrapper material developments, non-MA-bearing fuel performance and fabrication as well as preliminary knowledge performance and fabrication technologies of MA-bearing fuel have been collected and shared between the AF

[1] Although the carbide option was initially discarded of the fuel candidates, it has been introduced since 2008 regarding the significant and globally positive experience on carbide behavior under irradiation.

project members. This enables a review of the capability of the fuel and material candidates, the identification of the issues and the selection of the advanced fuel options from oxide, metal and nitride fuels.

III.A. Oxide fuel evaluation [3]

Oxide-based fuel has been the choice of most development and demonstration SFR programs worldwide in the past. Oxide fuels have thus reached an industrial maturity. The MOX fuel database is well established and collected data from Post Irradiation Examinations (PIE) have been used to develop and validate fuel performance codes such as GERMINAL (CEA) and CEDAR (JAEA). Despite two major drawbacks: poor thermal conductivity and chemical reactivity with sodium, the uranium plutonium oxide fuels show major advantages including high melting point (>2 400°C) and excellent stability under irradiation within a broad range of temperatures and burn-ups (up to 10 at%). Moreover, the low thermal conductivity presents some benefits, as the fuel center temperatures and radial thermal gradients under irradiation are high, enabling a high fission gas release fraction in the pin plenum (80% for a 10 at% burn-up) and thus a low overall fuel swelling (0.6-0.7%/at%). Regarding sodium-fuel reactivity, experimental studies as well as operational feedback from cladding rupture in experimental reactors have shown solutions to manage this drawback.

Regarding high burn-up capability of oxide fuels, FCMI seems to be manageable through the smeared density (which is defined as the ratio of the cross-sectional area of the as-fabricated fuel to the cross-sectional area defined by the cladding inner diameter). On the opposite, FCCI is a major issue with two inner cladding corrosion phenomena occurring at high burn-up, *i.e*. fuel cladding reaction and the fissile-fertile interface reaction. Fuel cladding reaction results in cladding wastage on a rather large area in the upper part of the fissile stack. The latter FCCI type affects a much more localized area located near the interface between fissile and fertile stacks. As these FCCI phenomena are partially linked to high Oxygen to Metal ratios of the fuel (O/M), low O/M ratios have to be investigated to reduce the cladding depletion. Thus fuel fabrication technology has to be adapted to provide the appropriate O/M.

Regarding presence of MA in the fuel, main issues include:

- fuel fabrication with high Am retention due to americium oxide volatility;
- MA addition effects on fuel properties and fuel microstructure (restructuring as well as redistribution of oxygen, plutonium and americium under irradiation);
- impact on FCCI;
- and fuel behavior in transients.

Some issues have partially been addressed with the preparation or the examination of MA bearing fuels irradiated within the frame of SUPERFACT-1 [4] and Am1 [5] experiments. SUPERFACT-1 performed in the 80's has provided the first results on the incineration of minor actinides in the homogeneous mode and has demonstrated the general good behavior of the MA-bearing fuel up to 6.5 at% at low linear power (~40 W.m^{-1}). The pellets were manufactured by sol-gel processes and were composed of solid solutions: $(U_{0.74}Pu_{0.24}Am_{0.02})O_2$ and $(U_{0.74}Pu_{0.24}Np_{0.02})O_2$. Post irradiation examinations have shown that the fuel underwent a similar microstructure evolution as standard fuels and neither Pu nor Am redistribution was found. Finally, a large helium release was measured in the americium bearing fuel (4 times greater than standard fuel). Additional irradiation data have been provided on Am (3 and 5 wt%) and Am/Np (2%/2%) bearing fuels by the Am1 irradiation test performed in Joyo: remote fabrication technology has been established at laboratory scale; out of pile measurements (melting point, O/M ratio, oxygen potentials, …) have been performed; PIE have shown that structural changes such as formation of lenticular pores and central void occurred within the first 10 minutes of the irradiation; no signs of fuel melting were found.

Finally, regarding the fabrication of MA bearing fuels for industrial applications, the

Powder Metallurgy standard process which generates dust has to be modified and/or simplified to limit the steps of powder handling. Prospects for the production of MA oxide fuels can be based on co precipitation or sol-gel methods. Pin manufacturing could be simplified too, using for instance a spherepac type process.

III.B. Metal fuel evaluation [6]

U-Zr and U-Pu-Zr alloy fuels with a sodium bonded fuel pin, were selected for many of the first SFR studies in the U.S.. Advantages of metallic fuels are their high thermal conductivity, high fissile density and available experience on metallurgy fabrication processes. An extensive database of their performances was generated [7], burn-ups of 10 at% at steady state were reached in reactors. Metal fuels capabilities have also been demonstrated up to 19 at% burn-up with Ferritic/Martensitic (F/M) steel clad. Moreover, metal fuels have been shown to exhibit sufficient margins to failure under transient conditions, despite their low melting temperature (<1 000°C).

One major drawback for U-Pu-Zr fuels is FCCI due to high fuel swelling especially at high burn-ups, which can lead to inter-diffusion phenomena between fuel and clad components, and then to the development of low temperature melting phases. As a consequence, fuel smeared densities and peak clad temperatures, have to be managed in order to prevent FCCI. FCMI seems not to be a major issue due to the high plasticity of metal fuels, if a large fuel-to-cladding gap as well as a large pin plenum, are available to accommodate fission gas. Other issues are sodium bond requirement and difficulties in modeling metal fuel behavior under irradiation due to the complexity of the U-Pu-Zr phase diagram.

The main issues for MA-bearing metal fuels are the same as for oxide fuels, with in addition, demonstration of:

- MA-bearing oxide feedstock reduction to metal alloy feedstock;
- an acceptable level of FCCI.

Some issues have already been partially addressed in the framework of the preparation or examination of MA bearing fuels, irradiated in the frame of X-501 [8], AFC1 [9] and METAPHIX-1 [10,11] experiments. The X-501 experiment has demonstrated the acceptable behavior up to ~6.5 at% of a U-20Pu-10Zr fuel containing 1.2 wt% Am and 1.3 wt% Np. The microstructure of the irradiated fuel is similar to U-Pu-Zr and FCCI of the HT-9 F/M clad is not strongly affected by small amounts of Am and Np. AFC1 metal compositions have shown excellent performance up to 8-10 at%. Non Destructive Examinations of METAPHIX-1 have shown neither crucial damage nor excessive deformation for metal fuel rods containing 5wt% or less MA, irradiated up to 2.5at%. In contrast, these experiments have shown that mechanical and thermal properties have not been seriously degraded by the addition of MA elements.

Finally, regarding fabrication of MA bearing fuel for industrial applications, a promising technique could be precision injection casting, which would provide a fuel slug without textured structure and which require a relatively short fabrication sequence easy-to-build and easy-to-use equipment.

III.C. Nitride fuel evaluation

Nitride fuels were identified as candidates for SFR, nearly three decades back, on the basis of their attractive physical and chemical properties *e.g.* a high heavy metal density, a strong thermal conductivity connected with a high melting temperature (>2 700°C) as well as a good compatibility with stainless steels, sodium, water (T≤60°), air (T≤25°C) and hydro-reprocessing. So, improved performances of nitride fuelled core (in comparison to oxide) such as a larger breeding ratio, higher linear heat rates and an improved safety have been expected. Nevertheless, fuel dissociation of (U,Pu)N fuels at a temperature substantially lower (~1 730°C) than the melting point if nitrogen overpressure is not maintained, has been identified as a critical issue in case of severe accidents. Another key issue consists of the ^{15}N enrichment requirement to prevent mostly the generation of the radiotoxic long lived ^{14}C from $^{14}N(n,p)$ reaction and additional He production from $^{14}N(n,\alpha)$ reaction.

The worldwide experience on nitride fuels has been limited to 150-200 irradiated pins for maximum burn-ups of 9 at% and linear heat rates of 45-130 $W.m^{-1}$. The overall swelling of nitride fuels fits a linear rate of 1.1 %/at% below a critical temperature which decreases from ~1 200 to ~950°C with increasing burn-up. Beyond the critical temperature, the swelling rate increases exponentially before being restrained by the cladding. Because of the high swelling rate, fission gas release remains low (<50%) even at high burn-up.

For high burn-up applications, FCMI is the major issue to manage in order to prevent large clad deformation and clad breach. It could, nevertheless, be solved by acting on the smeared density (~70-80%) and by favouring fuel open porosity for gas release.

For MA bearing fuels, because of the limited thermal stability of MA nitrides, MA redistribution could occur for high linear power or during power/temperature excursions. Furthermore, moderate temperature fabrication techniques have to be found to prevent Am losses.

To partially address these issues, experimental data from irradiation experiments on (U,Pu)N fuels (NIMPHE-1, NIMPHE-2 [12] and BORA-BORA [13]) as well as on MA bearing fuels (AFC1 [14] and FUTURIX-FTA [15]), are under analysis.

III.D. core materials evaluation [16]

The extremely high flux of fast neutrons in a SFR core is a main source of damage to subassembly materials used for the pin cladding and wrapper tube. Increasing final burn-ups (~20 at%) and core outlet temperatures (550°C) in Generation IV SFRs, imply that fuel structures must support both extremely high irradiation doses (~200 dpa) and higher peak temperatures: ~580°C for the duct and 650-700°C for the cladding according to the fuel type.

Austenitic steels have excellent material properties at high temperature and an acceptable swelling capability up to ~160 dpa, satisfying the requirements for the cladding of the current SFR systems. To achieve a higher burn-up in the Generation IV SFR system, steels with superior swelling resistance characteristics need to be utilized. To this end, F/M steels have been considered as primary candidates for the cladding and duct materials. Past experience on these steels has shown an excellent swelling resistance up to 200 dpa. However, these steels don’t have sufficient creep strength above 650°C to meet GIF goals except for metal core designs. The HT9 F/M steel has then been selected as a promising candidate for ducts and metal fuel cladding.

To extend the range of F/M steels to temperatures well above 650°C, Oxide Dispersion Strengthened steels made of a fine distribution of oxide particles in a F/M steel are promising candidates because of their high temperature strength.

IV. CONCLUSION

International collaborative activities are performed on fuel and core material developments within the Advanced Fuel project for Generation IV SFR systems. As a first milestone of the project plan, the R&D outcomes of national and collaborative programs have been collected and shared between the AF project members in order to review the capability of oxide, metal and nitride fuels and core materials candidates, to identify the issues and select the viable options.

Based on historical experience and knowledge on fast fuel development, as well as specific fuel tests currently being conducted on MA bearing fuels, both oxide and metal fuels emerge as primary options to meet quickly the performance and the reliability goals of Generation IV SFR systems. As the irradiation performance database for nitride fuels is limited, even if their attractiveness is high, these fuels are at an early stage of development with longer term R&D activities still required.

The status of core materials such as cladding and duct has been reviewed. The promising candidates are F/M and ODS steels.

The next step for the AF-PMB consists in introducing carbide fuels in the assessment (2008-2009), gaining knowledge and solving issues regarding core materials, performance and fabrication technologies of MA bearing fuels, from national and collaborative programs. A primary selection of advanced fuel(s) is expected by 2010.

List of abbreviations

AF: Advanced Fuel

F/M: Ferritic / Martensitic

FCCI: Fuel-Cladding Chemical Interactions

FCMI: Fuel-Cladding Mechanical Interactions

MA: Minor Actinides

MOX: Mixed Oxide

O/M: Oxygen to Metal ratio

ODS: Oxide Dispersed Strengthened

SFR: Sodium Fast Reactor

References

1. Carmack, J., *et al.*, “SFR status and ongoing results: GACID project”, *Proc. of GLOBAL 2009,* Paris (France), Sept. 6-10 (2009).
2. T. Mizuno, D.C. Crawford, *et al.*, “Status of the Advanced Fuel Project for Sodium-Cooled Fast reactor as A Generation IV Nuclear System Energy”, *Proc. of GLOBAL 2005,* Tuskuba (Japan), Oct. 9-13 (2005).
3. Raison, Ph., T. Mizuno, *et al*., “Sodium Fast Reactor Fuel Evaluation: Oxide Fuels”, *Proc. of NFSM*, Anaheim (USA), June 9-12 (2008).
4. Martin, Ph., M. Tourasse *et al*., “Progress in fuels for fast reactor”, *Proc. of IEMPT-9*, Nîmes (France), Sept. 26-29 (2006).
5. Tanaka, K., M. Osaka *et al*., “Research and Development of Americium-containing mixed oxide fuel for fast reactors”, *Proc. of GLOBAL’07*, Boise (Idaho), p. 897, Sept. 9-13 (2007).
6. Carmack, J., D. Porter *et al*., “GIF SFR Advanced Fuel Project: Evaluation of Metallic Fuel”, *Proc. of NFSM*, Anaheim (California), June 9-12 (2008).
7. Crawford, D.C., D.C. Porter and S.L. Hayes, “Fuels for sodium-cooled fast reactors: US perspective”, *Jour. Nuc. Mat.*, 371, p. 201 (2007).
8. Meyer, M.K., S.L. Hayes, *et al*., “Fuel design for the US Accelerator Driven Transmutation System”, *Proc. of AccApp’01*, Reno (Nevada), Nov. 12-15 (2001).
9. Hilton, B.A., D.L. Porter and S.L. Hayes, “Post irradiation Examination of AFC1 metallic transmutation fuels at 8 at%”, *Proc. of summer American Nuclear Conference*, Reno (Nevada), June (2006).

10. Breton, L., M. Masson, *et al*., "Metaphix-1 non destructive post irradiation examinations in the irradiated elements cell at PHENIX", *Proc. of GLOBAL'07*, Boise (Idaho), Sep. 9-13 (2007).

11. Fromont, M., J. Lamontagne, *et al*., "Behaviour of Uranium-Plutonium mixed nitride and carbide fuels irradiated in Phenix", *Proc. of GLOBAL'05*, Tsukuba (Japan), Oct. 9-13 (2005).

12. Zabudko, L.M., V.M. Poplavsky, "Investigation of nitride fuels fast reactors in Russia", *Proc. of JAERI conference 2004-015* (2004).

13. Hilton, B., D. Porter, S. Hayes, "AFC-1 Transmutation fuels – Post Irradiation hot cell Examination 4 to 8 at% - Final report", INL/EXT-05-0785 rev.1 (2006).

14. Jaecki, P., S. Pillon, *et al*., "Update of the FUTURIX-FTA experiment", *Proc. of IEMPT-9*, Nîmes (France), Sept. 26-29 (2006).

15. Cheon, J.S., C.B. Lee, *et al*., "Sodium fast reaction fuel evaluation: core materials", *Proc. of NFSM*, Anaheim (USA), June 9-12 (2008).

Current Status of Global Actinide Cycle International Demonstration Project

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Abstract

The current status of the Global Actinide Cycle International Demonstration (GACID) Project, being conducted under the GIF/SFR System Arrangement signed on February 15, 2006, has been summarized. The Project Arrangement for GACID was signed on September 27, 2007 and the following activities are underway as a part of the five years' effective period of the Arrangement: MA-bearing fuel raw material preparation, fuel fabrication, material property measurement, precedent irradiation tests in Joyo, irradiation behavior modeling and licensing study for pin-scale irradiations in Monju and Joyo, and program planning for future bundle-scale demonstration in Monju.

I. INTRODUCTION

Recovering and recycling the Minor Actinides (MAs), such as Neptunium (Np), Americium (Am) and Curium (Cm), with conventional Uranium (U) and Plutonium (Pu) in the spent fuel is generally called 'Actinide Recycle' or 'TRU Recycle' and the research and development (R&D) activities for its future commercialization are underway in several nations.

Actinide recycle (TRU recycle) has a potential to reduce the geological repository burden of the high-level radioactive waste. Moreover, an idea is being proposed that actinide recycle can drastically reduce the potential radioactive hazard in a timeframe of over thousands of years, thus significantly contribute to enhance the public understanding and acceptance of the radioactive waste and fuel cycle. More than several millions of years will be necessary to reduce the potential radioactive hazard of the current vitrified high-level radioactive waste, which assumes only U and Pu recycling, to the same radioactive-hazard level of the original U ore. On the contrary, this time period can be shortened to several hundreds of years by the actinide recycling, as shown in Figure 1.

At the same time, nuclear fuel materials containing MAs have some difficulties of their accesses, because Am is a gamma-ray emitter, and Cm is a neutron emitter and also a heat source. Another viewpoint is being discussed that

CURRENT STATUS OF GLOBAL ACTINIDE CYCLE INTERNATIONAL DEMONSTRATION PROJECT

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At the same time, nuclear fuel materials containing MAs have some difficulties of their accesses, because Am is a gamma-ray emitter, and Cm is a neutron emitter and also a heat source. Another viewpoint is being discussed that the need for an inaccessible separation work of the chemically similar elements for pure Pu separation can contribute to the nuclear non-proliferation policy.

Background R&D activities for actinide recycle research in France, the United States of America (US) and Japan, which are participating in the GACID Project, will be briefly reviewed, at first. Then the Project Plan and the current status of the Project will be reviewed and summarized.

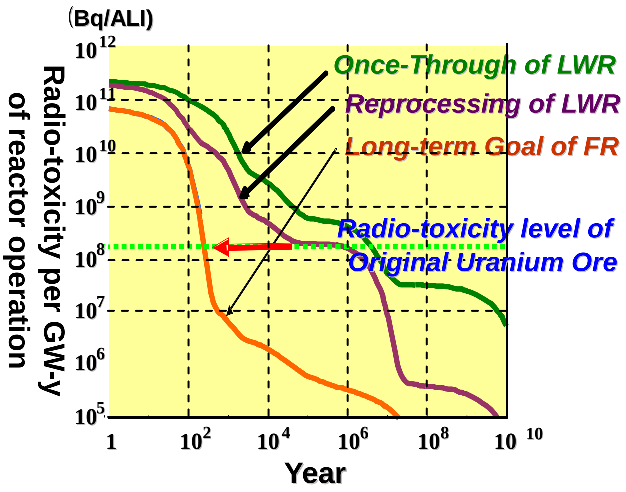

Figure 1: Potential Radioactive Hazard of High-level Radioactive Waste

II. BACKGROUND ACTIVITIES IN FRANCE[1]

The construction and operation experiences of a series of Fast Breeder Reactors (FBRs), such as Rapsodie (Thermal Power: 40 MWt, Initial Criticality: January 1967, Closed: January 1983), Phenix (Electric Power: 250MWe, Initial Criticality: August 1973, to be shutdown in 2009) and Super Phenix (Electric Power: 1 200 MWe, initial Criticality: September 1985, Closed: February 1998) have already been accumulated in France. These experiences will provide technical bases for the future sodium-cooled FBR development. Gas-cooled FBR is positioned in France as an alternative future option with the potential to combine the advantages of fast neutrons and possibly high temperature process heat generation.

As a result of French June 2006 Act on the sustainable management of waste, CEA is committed to evaluate by 2012 the industrial feasibility of MA actinide transmutation in GEN IV systems and in particular in SFR. For that, the various possible options for MAs recycling in a SFR will be evaluated. This includes the homogeneous recycling of MAs diluted in the standard fuel which is the objective of the GACID project, and the heterogeneous mode of recycling, in which MAs are concentrated in specific subassemblies (SA). For this option the preferred choice is to introduce them in a UO_2 matrix which means that MAs are recycled at core periphery in blanket SAs.

Basic pin-scale irradiation tests on MA-bearing Mixed Oxide (MOX) fuel have been carried out since the 1980's in France (SUPERFACT Program which still constitutes a reference experiment). Several experiments on MAs heterogeneous recycling were performed in Phenix this decade and have finished their irradiation now; their Post-Irradiation Examinations (PIEs) will bring interesting original results. Moreover, a radioactive-waste management scenario was proposed to the French Congress by the CEA in 2006, based on the Radioactive-Waste-Management Study Act.

The scenario fundamentally proposed is to separate the MAs form Fission Products (FPs) in the spent fuel, and to vitrify and to geologically reposit only the FPs. A concept is being proposed, as a first step, to construct a pilot-scale MA-fabrication plant in an existing commercial-based reprocessing plant for the engineering-scale demonstration of the MA fabrication technology from the high-level radioactive waste. Although this concept is a preliminary proposal, to produce a kg-order amount of MAs (Am in a first step) in this pilot-scale plant, to mix a part of it into a MOX fuel and to demonstrate the technical feasibility on an engineering scale by a bundle-scale MA-bearing fuel demonstration irradiation in an actual reactor, such as Monju, is being discussed, as shown in Figure 2.

In addition, the former President Chirac issued a communiqué in January 2006, saying that a Generation IV prototype reactor shall be commissioned in 2020 in France.

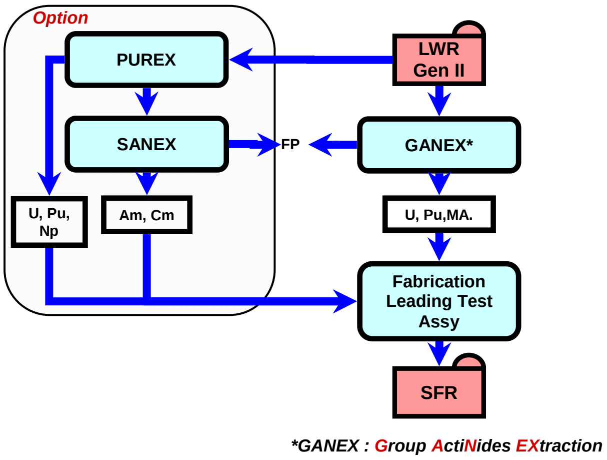

Figure 2: A Concept of MA-bearing Fuel Engineering-scale Demonstration in France

III. BACKGROUND ACTIVITIES IN THE U.S.A.[2]

The Generation IV International Forum (GIF) Project was established in July 2001, based on the activity of the US Department Of Energy (DOE) to promote the development of the next-generation reactor by international collaboration. This Project promotes the selection of the next-generation reactor concept in an international collaborative framework, based on the contributions of the participating nations.

At the same time, not only the reactor concept but also the corresponding next-generation reprocessing system is being developed by each participating nation. The Advanced Fuel Cycle Initiative (AFCI) Project is being conducted in the US to complement the GIF Project in parallel. The concept of the actinide recycle, based on the fast reactor system, is considered as one of the most promising candidates in the GIF Project due to its excellent advantages in natural-resource-utilization efficiency and environmental-burden reduction, that is sustainability, and proliferation resistance.

The AFCI program performs the research and development activities needed for a high proliferation resistant and advanced fuel recycling technology while minimizing the amount of the radioactive waste. A concept is being proposed, in parallel, to organize an international consortium by the fuel-supplier nations and to assure the fuel supply to the fuel-user nations, which commit the use of the nuclear power only for peace.

The actinide recycle scheme, which recovers and recycles the MAs together with U and Pu, is being assumed as a key candidate for advanced fuel recycling while maintaining a high proliferation resistance. Early demonstration of MA transmutation by MA-bearing fuel in actual reactors, such as Joyo or Monju, will help to promote the AFCI fuel cycle concept.

IV. BACKGROUND ACTIVITIES IN JAPAN[3]

The Fast Breeder Reactor Commer–cialization Strategic Study (FS) has been conducted in Japan since July 1997 and the final report was issued in July 2006 by Japan Atomic Energy Agency (JAEA) and Electric Power Utilities as a result of the Phase-II study. The TRU recycle (actinide recycle) scheme has been introduced from the viewpoints of environmental-burden reduction and proliferation-resistance enhancement, based on the FS's design requirements. Moreover, Low Decontamination (LD) fuel concept was being pursued, which allowed for residual FPs to simplify and to reduce the reprocessing procedures. This TRU and LD fuel concept is one of the reference concepts in the Fast Reactor Cycle Technology Development (FaCT) Project, which took over the FS Project.

The fabrication of such a TRU and LD fuel needs to be performed in a hot cell with sufficient radiation protection and heat removal, because of the MAs and FPs in the fresh fuel raw material.

For the commercialization of this MA-bearing fuel technology, engineering-scale pilot plants are to be constructed, and the technical feasibility of the reprocessing and fuel fabrication procedures should be demonstrated on an engineering scale. Six MA-bearing fuel pins have already been fabricated and irradiated in Joyo, and approximately 650 MOX fuel pellets, including MA-bearing fuel pellets, have already been sintered in a hot cell of JAEA by remote operation. However, these experiences are on an experimental scale and not sufficient for the commercialization of the technology. An engineering-scale demonstration of the reprocessing and fuel fabrication procedures is one of the issues to be resolved in the future.

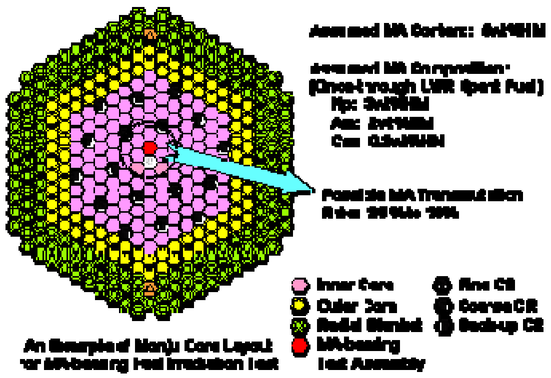

Figure 3: An Example of MA-bearing Fuel Irradiation Test Concept in Monju

At the same time, an engineering-scale irradiation demonstration of the MA-bearing fuel is also needed for the commercialization. Pin-scale experimental irradiations can be performed in Joyo, while the bundle-scale engineering demonstration irradiation is desired to be performed in a larger-scale reactor, such as Monju, as shown in Figure 3.

However the bundle-scale demonstration irradiation of MA-bearing fuel in Monju requires engineering-scale pilot plants for the reprocessing and fuel fabrication, which will need a certain period of time and budget for the preparation. International collaboration was considered to have the potential to reduce the necessary period of time for the bundle-scale demonstration.

On the other hand, Monju is the Japanese prototype FBR with an electric power of 280 MWe and its safety regulation is basically based on that of the current commercial-based electric-power-generation reactors. Precedent experimental data is needed by Joyo irradiation tests for the licensing in Monju.

Therefore a series of irradiation tests in Joyo and Monju was being planned to be discussed and succeeded to the GACID Project as shown below.

V. PROJECT ARRANGEMENT FOR GACID

The Project Arrangement (PA) for GACID was signed by the participating three Signatories, CEA France, USDOE and JAEA Japan on September 27, 2007, under the GIF Sodium-cooled Fast Reactors (SFR) System Arrangement signed on February 15, 2006. The discussions on the Project Plan were initiated in June 2004, in Tokyo among the specialists from CEA, USDOE and JAEA. Figure 4 shows the overview of the whole GACID Project conceptual scheme, as a result of the thorough discussions.

A series of irradiation tests in Joyo and Monju in three steps was proposed by JAEA and the Project Plan was determined to conduct the following irradiations.

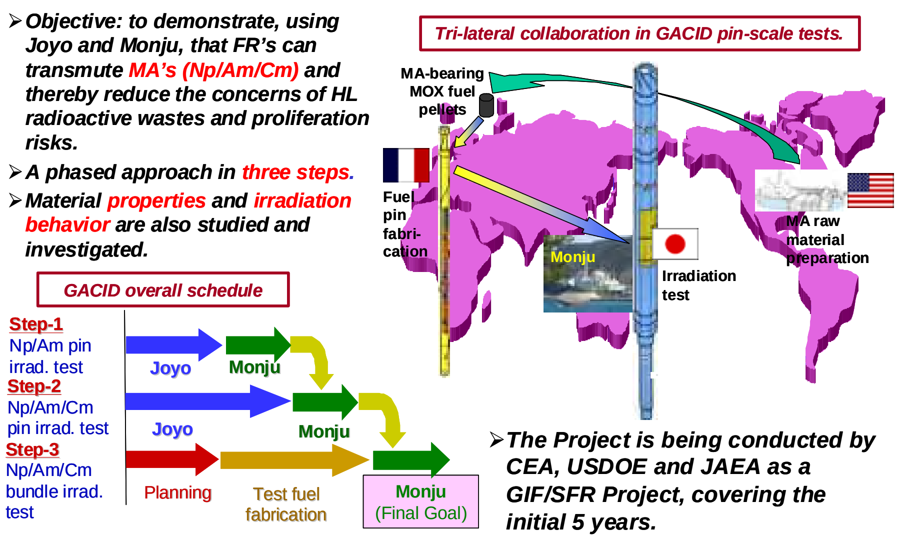

Figure 4: Overview of the GACID Project Conceptual Scheme

(1) Step-1: Precedent Limited MA-bearing Fuel Preparatory Irradiation Test

This test assumes Np-237 and Am-241 only as for the MAs. The radiation and heat source intensities of these isotopes are not so strong. Moreover only a single pin-scale irradiation test in Monju is planned, while the precedent Joyo irradiation tests are already underway. Therefore this test is expected to be implemented at an earliest stage of the Project, because the test fuel can be prepared by a minimum effort of only a small amount of the MA raw materials with minimal additional radiation protection for the bundle assembling. Although the MA isotopes are limited, the fundamental framework for the subsequent future MA-bearing fuel irradiation tests is expected to be established by this precedent test, as a model case of the irradiation tests in Monju.

All the necessary procedures for the MA-bearing fuel irradiation tests in Monju will be experienced as follows:

- MA raw material preparation and shipping.
- MA-bearing MOX fuel pellet sintering.
- Material property measurement and design correlation validation by measured data.
- Precedent Joyo irradiations and PIEs,
- Irradiation behavior modeling and design model validation by irradiation test data.
- Licensing in Monju.
- Test pellet and pin fabrication and shipping.
- Test bundle assembling and shipping, and
- Irradiation in Monju and PIE.

The basic geometries, dimensions and structures of the test assembly will be the same as the ordinary Monju driver fuel. Only the fuel composition of a single fuel pin in a bundle will be different. This concept will also contribute to the earliest implementation of the test.

(2) Step-2: Pin-scale Cm-bearing Fuel Irradiation Test

A full-range of MA composition is assumed for this test. Not only Np and Am but also Cm will be contained in the test fuel, although the test will be conducted on a pin scale. Gamma-ray and neutron radiation from Am-243 (From daughter nuclide: Np-239) and Cm-244 will no longer be ignored in this test. However only a single pin fabrication and irradiation will allow for the easier management of raw material preparation, radiation protection and heat removal issues. A precedent irradiation test in Joyo is being planned for the Monju irradiation licensing.

Fabrication and irradiation of Cm-bearing MOX fuel will be the world’s first trial.

(3) Step-3: Bundle-scale MA-bearing Fuel Irradiation Demonstration

After completing the above mentioned two steps of the precedent irradiation tests, the final goal, bundle-scale full-range-MA-bearing fuel irradiation demonstration, will be performed in Monju. Engineering-scale pilot plants for MA raw material preparation and MA-bearing fuel fabrication and assembling will be needed for this demonstration. Therefore the technical demonstration will be done in a reasonable time frame and the whole Project is to be conducted over a period of 20 years.

On the other hand, the effective period of time of the current PA is 5 years, with a first milestone two years after the PA signature to decide on the feasibility of pursuing the remaining tasks included in the original five years of the Project. Therefore the purpose of the current PA is to conduct collaborative R&D with a view to demonstrate the MA incineration capability of fast reactors on an engineering scale with a MA-bearing MOX fuel.

The schedule and allocations of activities during the initial 5 years of the Project are shown in Figures 5 and 6.

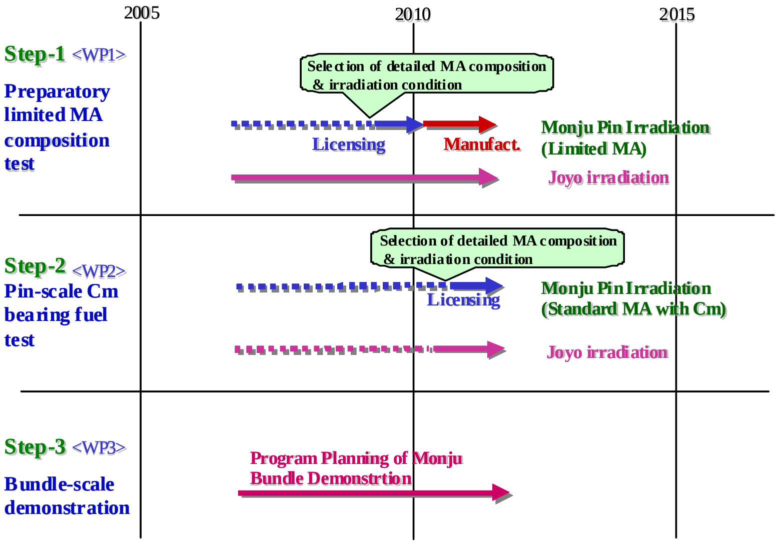

Figure 5: Schedule of GACID Project for Initial 5 Years

Organi-zation	*Individual Activities (5 years)*	*Common Activities*
JAEA (JP)	*- Precedent irradiation tests in Joyo (Am-1 test, etc.)* *- MA-bearing fuel material property measurement (low Am content fuel)* *- Licensing for Joyo and Monju irradiations* *- Preparation for Monju irradiation, PIE and transportation* *- Bundle assembling for Step-1 test fuel*	*- Project Management* *- MA bearing Fuel Material Property Data Evaluation* *- MA bearing Fuel Irradiation Behavior Modeling* *- Analysis and Evaluation of Irradiated Fuel Data* *- Program planning for Step 3 bundle-scale demonstration* *- Reflection to Gen IV SFR Design*
CEA (FR)	*- MA-bearing fuel material property measurement (high Am content fuel and Cm-bearing fuel)* *- Fabrication and transportation of Step-1 and -2 test fuel pins*	
DOE (US)	*- Raw material preparation for material property measurement* *- MA-bearing fuel material property measurement (supplemental data)* *- Raw Material Preparation and Transportation for Step-1 and -2 Test Fuel Pins*	

Figure 6: Allocations of Activities for Initial 5 Years

VI. CURRENT STATUS OF THE PROJECT

The Project is in progress based on the planned schedule and allocated activities mentioned above. The current status of the Project can be summarized as follows.

(1) MA Raw Material Preparation and Shipping

AmO_2 and NpO_2 feedstocks have been provided from USDOE to CEA/ ATALANTE at mid 2008 for Step-1 fuel material property measurement. Additional AmO_2 and CmO_2 feedstocks for the Step–2 fuel material property measurement are under preparation in USDOE.

(2) MA-bearing MOX Fuel Pellet Sintering

Am and Am/Np-bearing MOX fuel pellets have already been sintered and irradiated in Joyo in JAEA. Preliminary Am/Np-bearing MOX fuel sintering for material property measurement is underway in CEA and USDOE.

(3) Material Property Measurement

Material property measurement for Step-1 Am/Np-bearing MOX fuel is underway based on the planned measurement matrices of each organization. A Np content of up to 3wt%HM, Am content of up to 4wt%HM and Cm content of up to 0.6wt%HM is being assumed as the envelope MA composition of Once-through LWR, Recycled LWR (MOX) and Recycled FBR spent fuel with MA doping. Preliminary measurements in JAEA showed a tendency of slight decrease in melting point and deterioration in thermal conductivity, at lower temperature region, by Am doping for low-Am-bearing MOX fuel with an Am content of up to 3wt%HM.

(4) Precedent Joyo Irradiations and PIEs

Short term irradiations of 10 minutes and 24 hours for Am and Am/Np-bearing MOX fuel have already been completed in Joyo and the PIEs are underway. The preliminary PIE results showed an early restructuring of the pellets and Am redistribution behavior similar to Pu.

These results will be used for the irradiation behavior modeling for MA-bearing MOX fuel.

(5) Licensing in Monju and Joyo

The fuel specifications and licensing strategy for the Step-1 Monju irradiation test are being discussed. Discussions on the material-property and irradiation-behavior data-base preparation, linear heat rate, correlations, models and design methods for licensing, fabrication tolerances, etc. are underway. Similar discussions for the Step-2 Joyo irradiation test are also ongoing.

(6) Preliminary Program Planning for Bundle-scale Irradiation Demonstration

A notional overall schedule, and procedures and steps to achieve the bundle-scale MA-bearing fuel irradiation demonstration in Monju are under preparation.

VII. FUTURE PROSPECTS

The GACID Project is to be performed based on the current status, mentioned above, during the initial five years. The following issues are being identified to be resolved in the nearest future: fuel fabrication and characterization procedures, and material property measurement protocols. The details of the fuel fabrication and characterization procedures seem slightly different among the three participating organizations, although the basic procedures are the same. The material property measurement protocols also seem slightly different depending on the facilities and researchers to be used or assigned. These possible differences are to be investigated and harmonized so that the results will be technically consistent with each other. Moreover the preliminary program planning for the future bundle-scale demonstration is to be promoted. The final goal of the whole GACID Project is to be pursued.

At the same time, Joyo and Monju restart schedules are under discussion in JAEA. The results will be taken into account in the review of the Step-1 and Step-2 irradiation test schedules, together with the review of the material property measurement and test fuel preparation schedules, at the occasion of the PA review after two years of the effective period of the current PA, at the earliest.

VIII. CONCLUSION

The current status of the GACID Project has been reviewed and summarized together with the related background activities of each participating nation. Although each nation has individual future fuel cycle strategy, the final goal of the Project, bundle-scale engineering demonstration of MA-bearing fuel technology, is being fully shared. The Project will be conducted, as originally planned, until the end of the initial two years' effective period of the current PA, September 2009, first milestone to decide on the feasibility of pursuing or to review and revise the future plan, if necessary. Thus the current PA will be renewed and the Project will be continued until the end of the final goal of the whole GACID Project.

References

1. Pradel, Ph., "French Fuel Cycle Strategy and Partitioning and Transmutation Programme", *Proc. of Actinide and Fission Product Partitioning and Transmutation*", p. 27, OECD/NEA, Nimes, France, 25-29 September (2006).

2. Goldner, F., "The United States National Fuel Cycle Strategy", *Proc. of Actinide and Fission Product Partitioning and Transmutation*", p. 57, (ditto).

3. JAEA and JAPCO, Final Report on Fast Breeder Reactor Commercialization Strategic Feasibility Study Phase-II, JAEA and JAPCO, Oarai, Mito, July (2006).

SFR COMPONENT DESIGN AND BALANCE OF PLANT PROJECT

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I. INTRODUCTION

The SFR Component Design and Balance of Plant Project was formally initiated on October 11, 2007 with the signature of the Project Arrangement by the Commissariat à l'Énergie Atomique (CEA), the U.S. Department of Energy (DOE), the Japan Atomic Energy Agency (JAEA), and the Korea Atomic Energy Research Institute (KAERI). U.S. participants are Argonne National Laboratory (ANL) and Sandia National Laboratories (SNL). The main objective of the Project is to enhance the performance and economic competetiveness of Sodium-Cooled Fast Reactors (SFRs) through the development of advanced components and component-related technologies or through research and development of advanced energy conversion approaches such as the supercritical carbon dioxide (S-CO_2) Brayton cycle. In addition, the Project recognizes the significance of the experience that has been gained from SFR operation and upgrading in France, Japan, the Russian Federation, as well as the United States. This paper summarizes recent highlights of the Project with a focus upon the year 2008.

II. LESSONS LEARNED FROM SFR UPGRADING

CEA and JAEA have contributed the experience and lessons learned in upgrading PHÉNIX for improved safety and plant life extension through work carried out between November 1998 and early 2003, and upgrading JOYO to improve fast flux irradiation capabilities through work performed between October 30, 2000 and September 21, 2001.

The PHÉNIX experience encompassed in-service inspection and repair including under-sodium viewing by means of ultrasonic inspection of the conical shell supporting the reactor core, ultrasonic investigation of reactor vessel welds, upper hangers in the reactor vessel, primary and intermediate sodium circuits including the steam generators, and the fuel storage vessel, modification and repair of the steam generators involving cleaning using water of surfaces having residual sodium and reuse of drained sodium without caustic corrosion, replacement of the intermediate heat exchangers, changing (permutation) of a primary sodium pump, and changing (permutation) of control rod mechanisms as well as installation of an additional and new complementary shutdown system rod to assure reactor shutdown. The in-depth renovation of PHÉNIX demonstrated that the major technical operations were industrially feasible such as the ability to clean steam generators and reuse them afterwards and the ability to carry out ultrasonic investigations of the reactor vessel. Due to the success of the various operations, the operational life of PHÉNIX which commenced in 1974 was extended by ten years until March 2009

To increase the JOYO power level from 100 to 140 MWt, the intermediate heat exchangers, dump (sodium-to-air) heat exchangers, connecting sodium piping, and electric motors of the primary and secondary sodium pumps were replaced while maintaining a sodium level and fuel assemblies inside of the reactor vessel. In the particular instance of replacing the original sodium piping with new piping, work planning with the benefit of tests using full-size mockups, reduction of worker exposure time through training, installation of temporary shielding, and transparent seal bags were effective in reducing worker exposure and preventing the spread of contamination. Pipes were cut using a combination of initial bite cutting in an air atmosphere followed by roller press down cutting inside of a seal bag with measures to prevent foreign material (*i.e.*, small cut pieces or worker tools) from entering piping. Residual sodium was removed using cloths wetted with alcohol and water.

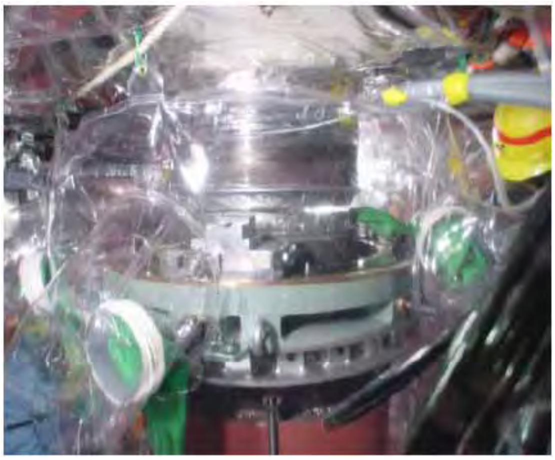

Figure 1: Seal Bag for Prevention of Oxygen Ingression at JOYO.

Experience and lessons learned from the PHÉNIX work were reported in a summary report containing an in-depth bibliography of reports by the participating organizations. The JOYO upgrading experience and lessons learned were reported in a summary report and two detailed JAEA reports in Japanese.

III. IN-SERVICE INSPECTION

CEA, JAEA, and KAERI have contributed information on the ongoing development of new and complementary in-service inspection technologies for in-vessel sodium components. JAEA is developing two ultrasonic sensors for under-sodium viewing. The first sensor is for real-time imaging to inspect for dislocations or deformations of structures. It is a piezoelectric element sensor that has a resolution of approximately 2 mm and supports an image processing time of approximately 0.5 second per image. The second sensor is for inspection to detect fatigue cracks. It is an optical diaphragm-type sensor and has a high resolution of approximately 0.3 mm. The sensors would be mounted on an under-sodium vehicle which would be driven or held on station in the sodium inside of a reactor vessel using six small magneto–hydrodynamic sodium pumps.

KAERI is developing a waveguide sensor approach enabling the ultrasonic transducer to be supported outside of the sodium pool and at a lower temperature at the reactor vessel upper head thereby minimizing the challenges to transducer performance and survival due to high sodium temperature, sodium chemical activity, and radiation from the nuclear core. The waveguide sensor is based upon the generation and transmission of Lamb waves (*i.e.*, surface waves propagating in an elastic solid) along a metallic strip waveguide. As shown in Figure 2, a 10 m long waveguide sensor module was fabricated incorporating from top to bottom a piezoelectric element ultrasonic transducer, a liquid wedge producing an A0 mode Lamb wave having a low frequency range below 2 MHz (Such zero-order Lamb wave modes exist over a range of frequencies and can transmit a significant amount of energy with low attenuation.), a waveguide strip plate surrounded by an acoustical shielding tube, and an emission face for the ultrasonic beam. The waveguide sensor approach has been demonstrated by feasibility experiments in water. The 10 m long waveguide sensor has a resolution of approximately 2 mm.

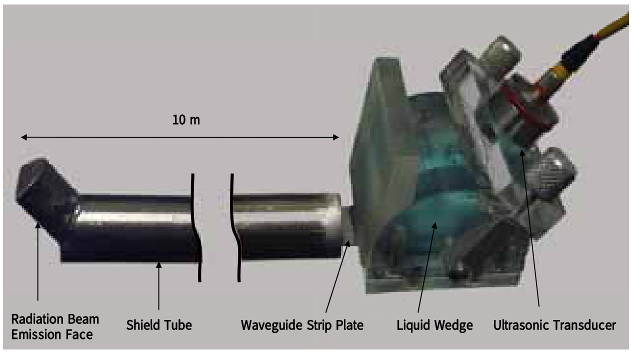

Figure 2: KAERI Waveguide Sensor Module.

CEA is developing an approach for inspection of in-vessel sodium components for dislocations or deformations using ultrasonic transducers strictly located outside of the reactor vessel.[1] The ultrasonic waves must thus propagate across several interfaces in traveling through the reactor vessel, potentially an intermediate structure, and finally the structure to be inspected. Initial development has focused upon a configuration involving three vertical steel walls for which two approaches have been investigated. The first employs ultrasonic waves at normal incidence using a single transducer and tuning the signal frequency and duration to promote constructive interference of the various reflected waves which is dependent upon the thicknesses and separation distances of the walls. The second approach utilizes oblique incidence and employs two transducers to take advantage of the existence of angular conditions for which transverse waves can have amplitudes exceeding those of longitudinal waves at normal incidence.

IV. LEAK-BEFORE-BREAK

KAERI is developing an evaluation approach for application of the leak-before-break principle to sodium piping and structures fabricated from Mod.9Cr-1Mo (G91) ferritic steel. To this end, KAERI has been performing creep-fatigue crack initiation and crack growth tests, fatigue crack growth tests, and creep crack growth tests on Mod.9Cr-1Mo tubular specimens including defects. Fatigue crack growth rates have been obtained for temperatures of 500, 550, and 600°C and for load ratios of 0.1 and 0.3, and the information was contributed to the Project. Test data will be used to develop high temperature defect assessment procedures.

V. SUPERCRITICAL CO_2 BRAYTON CYCLE

ANL, SNL, JAEA, and KAERI have contributed results from ongoing work covering complementary facets of the development of the supercritical carbon dioxide (S-CO_2) Brayton cycle for advanced energy conversion for SFRs. Test results have been contributed on the performance testing of small-scale diffusion-bonded heat exchangers representative of compact heat exchangers having cores similar to those envisioned for use as recuperators in the S-CO_2 cycle, testing of a small-scale S-CO_2 compressor, experiments on sodium-CO_2 interactions, and CO_2 corrosion and carburization tests.

ANL provided results from performance testing of a small-scale 17.5 KW nominal heat duty Printed Circuit Heat Exchanger™ (PCHE™, Heatric Division of Meggitt (UK) Ltd.) for CO_2-to-CO_2 heat exchange under prototypical low temperature recuperator (LTR) conditions of pressure, temperature, and scaled flowrate. The ANL S-CO_2 Heat Exchanger Testing Facility was modified from its earlier configuration for CO_2-to-water heat exchange testing to a new configuration incorporating a low pressure S-CO_2 loop with electrical resistance heating and a separate high pressure S-CO_2 loop having a different flowrate with heat rejection to water. The two loops are thermally connected through the PCHE which has a core mocking up a portion of the core of a full-size LTR module. Data was obtained for sixty-three sets of steady state operating conditions for which heat exchange rates and pressure drops were determined. Friction factor and heat transfer correlations for zigzagged semicircular micro–channels[2] were tested against the data.

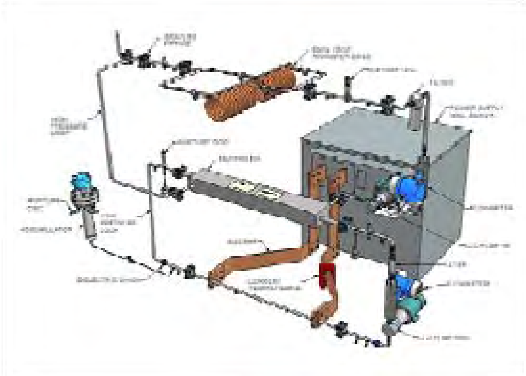

Figure 3: ANL PCHE™ Performance Tests for Prototypical Low Temperature Recuperator Conditions.

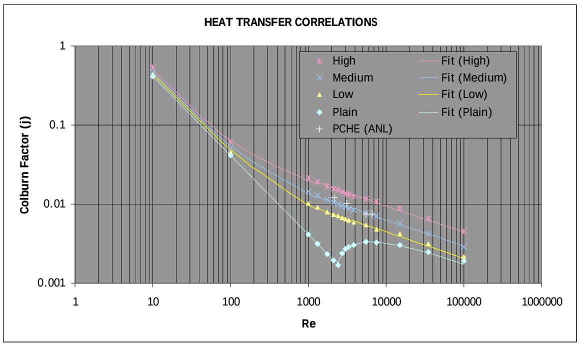

Figure 4: Comparison of ANL PCHE™ Performance Data with Correlations for Heat Transfer in Zigzagged Semicircular Channels.[2]

JAEA provided results on the heat exchange and pressure drop performance of small-scale compact diffusion-bonded heat exchangers (high and low temperature recuperators), compressor efficiency near the critical point, and flow stability near the critical point from tests carried out in a closed 10 MPa S-CO_2 loop at JAEA incorporating an electrical heater, piston CO_2 compressors, heat exchangers with cores simulating portions of the cores of high and low temperature recuperators, and CO_2 expansion through a valve in place of a turbine. Heat exchangers incorporating either zigzagged semicircular channels or a new configuration with S-shaped fins and interconnected channels were tested. Data showed that the S-fin recuperator has lower pressure drops than a zigzagged fin recuperator while both provide the same heat transfer performance. The tests also confirmed that compressor efficiency increases near critical point and that the loop was not subject to CO_2 flow instability.

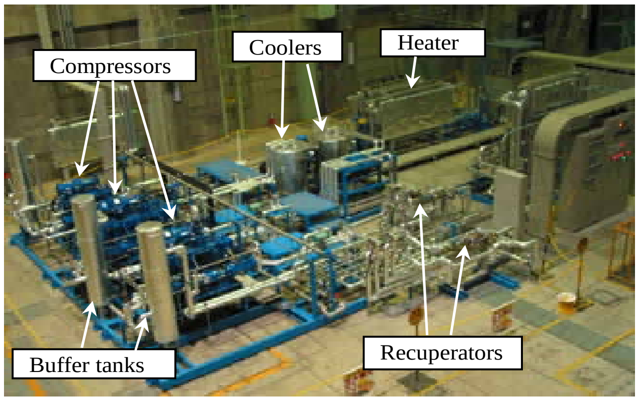

Figure 5: JAEA 10 MPa Small-Scale S-CO_2 Loop.

KAERI has developed a new compact diffusion-bonded heat exchanger design incorporating airfoil-shaped fins separating interconnected channels as opposed to non-connected independent zigzagged channels. Calculations with the FLUENT computational fluid dynamics code showed that the heat exchanger design with airfoil shapes achieves the same heat exchange rate while significantly reducing the pressure drops. Small-scale heat exchangers incorporating the new airfoil-shaped design and zigzagged semicircular channels were fabricated through diffusion bonding. The heat exchangers shall be tested in a new S-CO_2 HEX testing facility at KAERI.

SNL has been coordinating the design, construction, and testing of a small-scale (~ 1 MWt) S-CO_2 Brayton cycle power conversion system to confirm the performance of S-CO_2 cycles and demonstrate key technical issues for the cycle. SNL provided information on the fabrication and initial results from operation and testing of a single 50.2 KW centrifugal compressor in a small-scale S-CO_2 compression loop operating near the CO_2 critical point. The compressor is driven by a motor/alternator; heat generated during compres-sion is removed in a waste heat chiller. The compressed CO_2 is expanded after flowing through an orifice simulating the expansion in a turbine. Tests were conducted covering compressor inlet conditions at the critical point, on the liquid side of the two-phase dome, on the vapor side of the dome, and inside the dome. The results confirm that the compressor can be operated and that the S-CO_2 cycle should be controllable near the critical point. Information was also provided on the design and construction of a split flow S-CO_2 recompression loop incorporating the main compressor together with a second recompressing compressor.

JAEA and KAERI contributed results from sodium-CO_2 interaction tests. Chemical reaction tests were carried out at JAEA by contacting a small pool of sodium with overlying CO_2 gas.[3] Continuous reactions between sodium and CO_2 accompanied by flames occurred at temperatures higher than 570 to 580°C. This threshold exceeds the core outlet temperature in SFR designs. The main reaction products were determined to be Na_2CO_3 and gaseous CO; the heat of reaction was measured as 50 to 75 KJ/mol of sodium. KAERI has constructed a new apparatus to investigate sodium-CO_2 interactions in both a sodium pool configuration with CO_2 above the sodium pool (*i.e.*, surface reaction tests) and a vertical cylindrical capsule in which CO_2 is injected near the bottom of a small column of sodium (CO_2 injection tests). Some tests were conducted in the pool configuration up to 600°C at 0.1 MPa exhibiting a reaction threshold temperature of 510°C above which the reaction occurs much more rapidly, similar to the threshold effect at 580°C in the JAEA tests. KAERI is analyzing the results to determine reaction rates.

Figure 6: KAERI Sodium-CO_2 Reaction Test Facility

CEA shall also perform sodium-CO_2 interaction tests in which a CO_2 jet is directly injected inside of a 2 Liter sodium pool in the DISCO2 (Determination of Sodium-CO_2 Interactions) facility. Local temperatures will be measured with a movable comb of thermo-couples (Figure 8). Results will be utilized in adjusting existing modeling for sodium-water reaction kinetics[4] for application to sodium-CO_2 reactions. Tests will begin during 2009 and shall be reported to the Project.

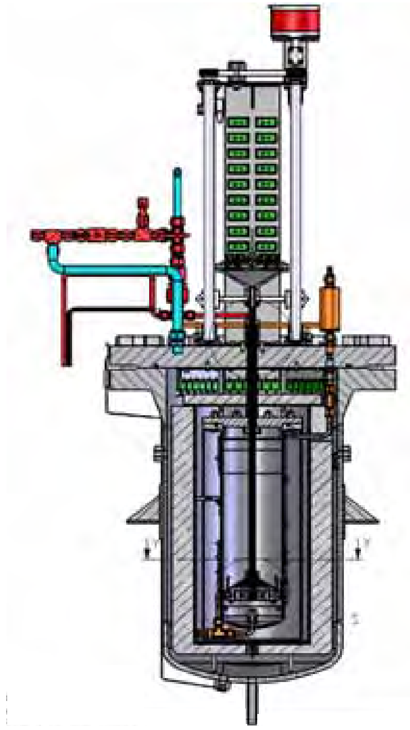

Figure 7: CEA DISCO2 Sodium-CO_2 Interaction Facility.

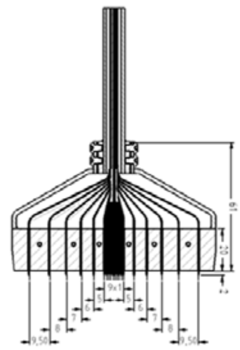

Figure 8: Comb of Thermocouples Inside of the DISCO2 Vessel Moved by a Stepping Motor.

JAEA contributed data on CO_2 corrosion and carburization of 12 Cr martensitic steel and the Japanese fast reactor stainless steel, 316FR, in flowing CO_2 at 10 MPa. No breakaway phenomena were observed for either material in 5 000 hour tests which confirmed good corrosion resistance for the stainless steel material.

The corrosion of 12Cr steel versus time followed a parabolic curve. Similar tests are in progress at CEA.

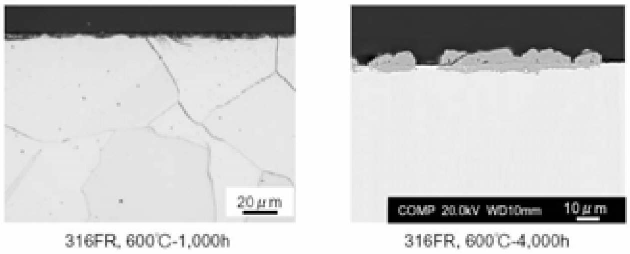

Figure 9: Micrographs from JAEA CO_2 Stainless Steel Oxidation and Carburization Tests.

ANL, JAEA, and KAERI contributed analyses of the behavior of SFRs incorporating S-CO_2 Brayton cycle power converters. The ANL Plant Dynamics Code for system level transient analysis of a SFR with a S-CO_2 Brayton cycle power converter was used to calculate the cycle behavior following a reactor scram in the 96 MWe (250 MWt) Advanced Burner Test Reactor (ABTR) SFR concept.[5] An interval of 400 seconds is required for the primary sodium coolant flow to transition to natural circulation following receipt of the scram signal resulting in tripping of the primary sodium pumps and disconnection of the generator from the electrical power grid. The S-CO_2 cycle is calculated to continue to remove heat from the reactor at a diminishing rate via the intermediate sodium circuit over the 400 seconds. Power continues to be generated in the turbine which spins the compressors which are installed on a common shaft while heat is rejected in the cooler. However, the cycle pressures and temperatures decrease during this time such that the minimum cycle pressure and temperature are calculated to fall below the critical values. The calculation shows that there is a window of 400 seconds for startup of the normal shutdown heat removal system incorporating a shutdown heat removal S-CO_2 pump and cooler. During this window, the S-CO_2 cycle continues to cool the reactor. CEA, ANL, and SNL have recently initiated a new collaboration under the Project which includes the creation of a postdoctoral position at CEA Cadarache involving work with the Plant Dynamics Code.

JAEA contributed a preliminary concept for a SFR with a S-CO_2 Brayton cycle power converter[6] in which the intermediate sodium circuit is eliminated.[7] The resulting plant efficiency is approximately 42% and the volume of the reactor building is reduced by 20% by adopting the S-CO_2 cycle and eliminating the intermediate sodium circuit. Both a helical coil tube sodium-to-CO_2 heat exchanger and a compact diffusion-bonded sodium-to-CO_2 heat exchanger were designed for the SFR. As part of the safety evaluation of a sodium-CO_2 reaction event, calculations were performed for a postulated double-ended guillotine rupture failure of one tube of the helical coil sodium-to-CO_2 heat exchanger installed in the primary sodium circuit.[8] The calculated maximum pressure in the primary sodium circuit resulting from the release of CO_2 is 0.28 MPa which does not threaten the primary circuit structural integrity. A voiding reactivity due to gas in the core is calculated to reach 0.046 $ which has no significant effect upon core safety.

KAERI has developed the STASCOR computer code modeling the chemical reactions between sodium and CO_2 in a flowing sodium circuit. The long-term behavior following a postulated tube rupture was evaluated for a shell-and-tube type sodium-to-CO_2 heat exchanger in the KALIMER-600 design.

VI. CONCLUSION

The SFR Component Design and Balance of Plant Project is facilitating the fruitful exchange of information and establishment of collaborations mutually beneficial to all participants. The lessons learned during upgrading of PHÉNIX and JOYO are of great significance and benefit all members of the Project. The research and development of in-service inspection approaches is following three parallel paths each of which is highly innovative in its own right. Significant improvements or breakthroughs in the ability to perform in-service inspection of in-vessel sodium components may result from this ongoing work. Data needed for the evaluation of leak-before-break for Mod.9Cr-1Mo ferritic steel sodium piping and components is systematically being generated. Finally, the Project is making highly significant contributions to the development and demonstration of S-CO_2 Brayton cycle advanced energy conversion spanning the development and performance testing of compact heat exchangers, development and testing of small-scale S-CO_2 turbomachinery and a complete integrated cycle, sodium-CO_2 interaction testing, CO_2 oxidation and carburi–zation tests, as well as the analysis of system behavior for SFRs incorporating S-CO_2 Brayton cycle power converters.

Acknowledgements

The reported work represents the efforts of over sixty individuals at CEA Cadarache, CEA Marcoule, LCND Laboratory (Aix en Provence), ANL, Kansas State University, SNL, JAEA, Tokyo Institute of Technology, Mitsubishi Heavy Industries, and KAERI.

References

1. Baque, F., C. Lhuillier, G. Gobillot, P. Brau, M.A. Ploix and J.M. Augem, “Ultrasonic Techniques for Improving Inspection of Sodium Cooled Systems,” Paper 80, *Proceedings of ANIMMA ‘09,* Marseille, France, June 7-10, 2009.

2. Southall, D., R. Le Pierres and S.J. Dewson, “Design Considerations for Compact Heat Exchangers,” Paper 8009, *Proceedings of ICAPP ‘08,* Anaheim, California, June 8-12, 2008.

3. Ishikawa, H., S. Miyahara and Y. Yoshizawa, “Experimental Study of Sodium-Carbon Dioxide Reaction,” *Proceedings of ICAPP ‘05*, Seoul, Korea, May 15-19, 2005.

4. Gicquel, L., P. Hobbes, A. Saboni, N. Simon and C. Latge, “Modeling of Supercritical Carbon Dioxide Jets in Liquid Sodium,” *Proceedings of SFGP ‘09,* Marseille, France, October 14-16, 2009.

5. Sienicki, J.J., A. Moisseytsev, D.H. Cho, Y. Momozaki, D.J. Kilsdonk, R. Haglund, C.B. Reed and M.T. Farmer, “Supercritical Carbon Dioxide Brayton Cycle Energy Conversion for Sodium-Cooled Fast Reactors/Advanced Burner Reactors,” *GLOBAL 2007, Advanced Nuclear Fuel Cycles and Systems*, Boise, Idaho, September 9-13, 2007.

6. Muto, Y. and Y. Kato, “Design of Turbomachinery for the Supercritical CO_2 Gas Turbine Fast Reactor,” *Proceedings of ICAPP ‘06*, Reno, Nevada, June 4-8, 2006.

7. Mito, M., N. Yoshioka, Y. Ohkubo, N. Tsuzuki, and Y. Kato, “Fast Reactor with Indirect Cycle System of Supercritical CO_2 Gas Turbine Plant,” *Proceedings of ICAPP ‘06,* Reno, Nevada, June 4-8, 2006.

8. Ohyama, K., M. Kishida, M. Mito, N. Yoshioka, and Y. Kato, “Design Study of CO_2-Na Reaction Events of Super-critical CO_2 Indirect Cycle Gas Turbine Fast Reactor,” *Proceedings of ICAPP ‘07,* Nice Acropolis, Nice, France, May 13-18, 2007.

CURRENT STATUS AND PROSPECTS OF R&D ON GENERATION IV SFR SAFETY AND OPERATION PROJECT

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I. INTRODUCTION

Generation IV Nuclear Energy Systems are being developed under the initiative of Generation IV International Forum (GIF) begun in 2000. The SFR was selected as one of the promising concepts together with other five concepts.[1] The System Arrangement for the International Research of 15 February 2006 constitutes a framework to carry out the research and development work necessary to establish the viability and to optimize the performance of the GIF SFR, and to facilitate (but not undertake) the eventual demonstration of the SFR system. For the purposes of coordinating the collaborative R&D among the member countries, the Safety and Operation Project Management Board (SOPMB) was organized under the SFR SSC based on the SFR System Arrangement. The member countries of SOPMB are France, Japan, Republic of Korea, and the United States of America. The Project Arrangement was concluded in June 2009 for the implementation of collaborative R&D. This project includes 1) analyses and experiments that support approaches and assess performance of specific safety features, 2) development and verification of computational tools and validation of models employed in safety assessment and facility licensing and 3) acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants. This paper describes the current status and prospects of R&D on SFR safety and operation project.

II. GOALS AND DEVELOPMENT TARGETS RELATED TO SAFETY AND OPERATION

II.A General Goals for Generation IV Nuclear Energy Systems Related to Safety and Operation

Three goals for the Generation IV nuclear systems have been defined in the safety and reliability as listed below.[1]

- *Safety and Reliability-1, Generation IV nuclear energy systems operations will excel in safety and reliability.*
- *Safety and Reliability-2, Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*
- *Safety and Reliability-3, Generation IV nuclear energy systems will eliminate the need for offsite emergency response.*

II.B Key SFR Development Targets on Safety

To effectively meet the Generation IV systems goals, the SFR R&D plan will focus on particular technology development efforts. Critical issues for the SFR technology are examined for achieving the enhancement of safety. Some key development targets for the SFR are summarized below.

With regard to reactor safety, technology development efforts focus on two general areas: assurance of passive safety response, and techniques for evaluation of bounding events. Advanced SFR designs exploit passive safety features to increase safety margins and to enhance reliability.

The system behavior will vary depending on system size, design features, and fuel type. R&D for passive safety will investigate phenomena such as axial fuel expansion and radial core expansion, and design features such as Self-Actuated Shutdown Systems (SASS) and passive decay heat removal systems. The ability to measure and verify the performance of these passive features must be demonstrated. Associated R&D will be required to identify bounding events for specific designs and investigate the fundamental phenomena necessary to prevent severe accident progression.

The favorable passive safety behavior of fast reactors is expected to reduce the probability of severe accidents with potential for core damage. Nevertheless, design measures to mitigate the consequences of severe accidents are being considered. This approach is consistent with the defense-in-depth philosophy of providing additional safety margin beyond the design basis. A common safety approach incorporating the physical and chemical characteristics of the materials handled in the reactor (chemical activity and radio-toxicity, etc.) and unique SFR design features and phenomena should be established. The goal is to render the risk of installing SFR systems much lower than the risk of energy alternatives. Achieving this level of safety should result in licensing and regulatory simplifications that may in turn result in reduced system cost. To do this, probabilistic safety evaluations will be needed to identify design tradeoffs that assure very high levels of public health and safety.

III. TECHNOLOGICAL SAFETY ISSUES INDENTIFIED BY GENERATION IV SFR DESIGNS

Three reactor systems have been proposed as options for Generation IV SFR system. These concepts are based on the significant knowledge and experience accumulated so far, but they also adopt innovative technologies.

- A large size (600 to 1 500 MWe) loop-type sodium-cooled reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.[2]
- A medium or large size (600 to 1 500 MWe) pool-type system also supported by a fuel cycle.[3]
- A small size (50 to 150 MWe) modular-type sodium-cooled reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.[4]

Assurance of passive safety response of a medium or large size pool-type SFR and a small size modular-type SFR in bounding events or Anticipated Transient Without Scram (ATWS) is one of the predominant developmental issues. For accurate predictions of the passive response of the reactor, it is necessary to develop advanced modeling of the transient thermal-hydraulic, and mechanical behavior of the reactor components that are the basis for inherent reactivity feedback estimates. To enhance the reactivity feedback models, an accurate three-dimensional core thermal-hydraulic model should be developed and fully linked to the reactor physics model. Validation of these models is an integral part of the development process.

The characteristics of metallic fuel behavior in a medium or large size pool-type SFR and a small size modular-type SFR during a hypothetical core disruption accident are an issue to be studied to ensure the expected low probability of the CDA occurrence in a metal fuel core and to prepare an adequate design for managing severe accidents.

The liquidus, solidus, and mobilization temperatures of metallic fuel and steel mixtures will be investigated. Molten fuel relocation behavior is also an important research area for the development of the predictive models.

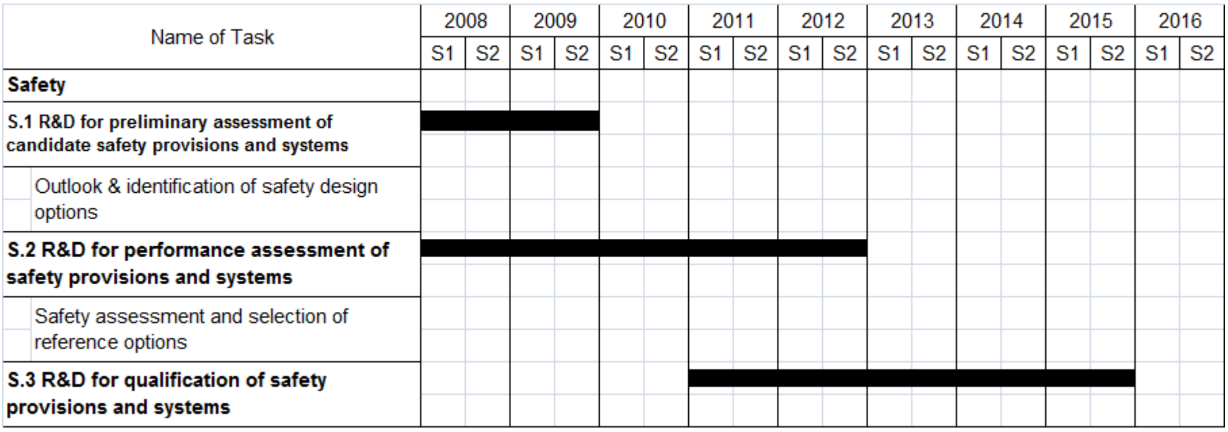

Figure 1: Safety Work Package Schedule.

In the safety approach for a large size loop-type SFR with mixed uranium-plutonium oxide fuel, the technology gaps center around two general areas: assurance of passive safety response, and the technology for evaluation of bounding events. With appropriate design features, passive safety response can be assured in large size reactors, and the plant utilizes passive safety measures to increase its reliability. R&D for passive safety will investigate relevant thermal-hydraulic and mechanical phenomena and design features such as self-actuated shutdown system and passive decay heat removal systems. R&D for bounding events will investigate the fundamental phenomena and design features for severe accident consequence mitigation. One approach is to introduce design measures to enhance the fuel discharge. The EAGLE[5] project is aimed at demonstrating the effectiveness of inner duct structure to enhance the fuel discharge without the formation of large molten fuel pool and to obtain an insight for the prevention of severe re-criticality in a sodium-cooled MOX fuel core.

IV. OUTLINES OF SAFETY AND OPERATION PROJECT

IV.A Safety

The safety design and its assessment must be coordinated with the overall design activity.

In particular, the time schedule of safety R&D work must also take place in concert with corresponding design work.

As shown in Figure 1, the safety project includes three stages corresponding to three work packages (WPs): (S1) R&D for preliminary assessment of candidate safety provisions and systems, (S2) R&D for performance assessment of safety provisions and systems, and (S3) R&D for qualification of safety provisions and systems. Each stage includes R&D of passive/active safety issues, severe accident issues and framework and methods of safety architectures that are necessary to support the system integration and assessment.

Work package S1 provides preliminary assessment of candidate safety provisions and systems introducing innovation. Because innovative and/or new design features will be adopted, critical issues in those features should be reviewed from the viewpoint of safety. Innovative/new features include new types of fuel such as recycle transuranic fuels (oxide, metallic, nitride, and carbide), and new types of plant designs aiming at economic advantages. Various passive/active safety features are considered in the designs, such as the SASS, the Passive Decay Heat Removal Circuit (PDRC), and the inherent shutdown and natural circulation shutdown heat removal features. Passive/active safety analysis tools will be developed. The feasibility of the innovative safety features will also be reviewed, and severe accident manage–ment measures will be assessed. Results of severe accident tests for MOX fuel and inherent safety tests for metallic fuel will be utilized for safety review.

Work package S2 provides performance assessment of safety provisions and systems in order to evaluate whether the design meets the safety requirements. Various R&D activities are planned to prepare the analysis tools needed for the performance assessment of the safety design options. Passive safety analysis tools will be validated through Joyo ATWS simulation tests. Additional results of severe accident tests for MOX fuel will be obtained and used for the validation of the models in severe accident analysis tools, including the SIMMER code. Post accident heat removal and in-vessel retention for metallic fuel systems will be investigated and assessed to establish Post Accident Materials Relocation (PAMR)/Post Accident Heat Removal (PAHR) scenarios. All of these R&D results will be used for establishing accident scenarios in the performance assessment of the design options. And a performance assessment for the proposed design of the safety architectures will be implemented.

Work package S3 provides a technical basis for safety assessment aiming at design optimization. A safety assessment of the reference designs will be interactively continued in order to refine the designs. For this purpose, further R&D for qualification of passive/active safety and severe accident analysis tools are considered, and the results will be reflected in the safety assessment.

IV. B Reactor Operation and Technology Testing

Operation technologies, experiments/ testing for computational tools validation, and demonstration of innovative technologies by using existing SFR plants are essential for SFR design and technology viability demonstration. This project is aimed at gathering the contributions from existing reactors to SFR design. Such contributions include: data acquisition, code validation, operation feedback (on safety, operation and maintenance), and innovative technology testing. The time schedule of Reactor Operation and Technology Testing (ROTT) project is shown in Figure 2. Additional ROTT tasks should be added later as the SFR project makes progress. (Utilization of Monju start-up within this testing program will be discussed when the schedule for these events become better determined.)

Tasks related to System Integration & Assessment (SIA) project are performed in Work package Op.1 in order to acquire validation data through the Phenix operation period that includes end-of-life tests and lifetime extension project, Monju start-up tests, and decommissioning of SPX. Plant behavior experiments will be performed at Phenix at its end-of-life time. Test data of thermal hydraulics including natural convection, neutronics, and investigation of negative reactivity transient will be used for validation of the MARS-LMR and SSC-K codes. Thermal-hydraulic data from Monju start-up tests will be used to verify and validate both computer models and design capabilities. Testing and validation could include design capabilities such as natural convection. The CEA CATHARE-ML general system code for a loop type reactor will be validated by using Monju transient recordings during commissioning tests and operation.

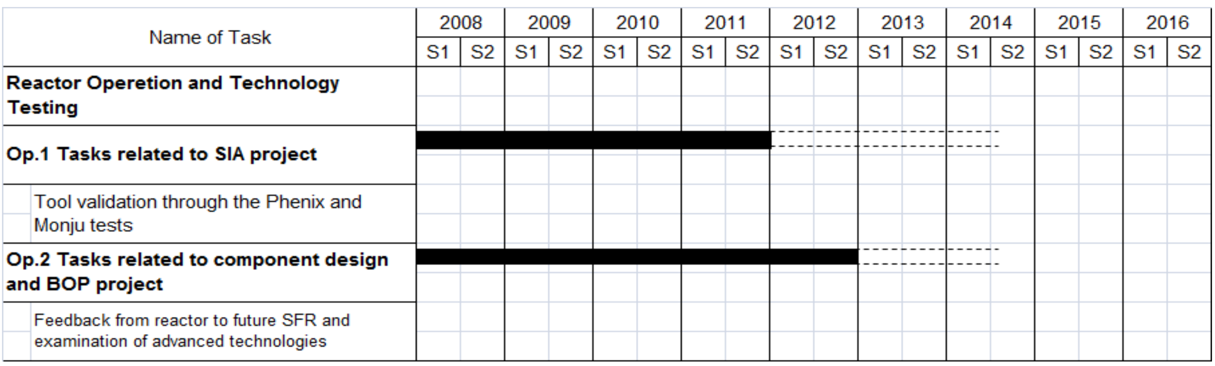

Figure 2: Reactor Operation Technology Testing Work Project Schedule.

The US will share the results from validation of new fuel assembly thermal hydraulic models with EBR-II Shutdown Heat Removal Test (SHRT) data[6]. The analyses will employ detailed sub–channel temperature measurements from the XX09 instrumented fuel assembly. The new models are being implemented in multiple computer code frameworks and solution architectures that include traditional subchannel modeling, commercial CFD modeling, and developmental advanced simulation capabilities. In order to contribute to a better design of future SFRs, the knowledge of decommissioning obtained from LMFR in the aspects of both technology and economics will be accumulated for integration and feedback.

Tasks related to Component Design and Balance-Of-Plant (BOP) project are performed in Work package Op.2 in order to issue recommendations for future SFRs in the areas of In-Service Inspection (ISI) methodology, sodium quality control and monitoring, radioactive impurities behavior, and tritium transfer based on the experience of existing reactors. Evaluation of ISI methodology will be based on results of its application to existing reactors. Analysis of the coolant chemistry operating feedback from sodium circuits in Phenix and Monju will be carried out during start-up and subsequently in steady-state. Additionally, tritium measurements at the Monju reactor will be analyzed to improve understanding of the reactor's tritium source and its subsequent transfer and accumulation on the reactor system and components. The improved understanding of radioactive material transport (activated corrosion products, fission products) and other behaviors in the primary circuit of an SFR that results from these activities will help develop design requirements for impurities control, oxygen control, and sodium purification system in future SFRs.

V. DETAILED DESCRIPTION ON SOME TOPICS IN THE SO PROJECT

In this section some topics in the SO Project are described in more detail.

In the US, research activities sponsored by the U.S. Department of Energy focus on fulfilling the Generation IV safety and reliability goals with a technology development program that includes demonstrative reactor concept safety performance evaluations. These analyses employ models validated with experimental data, and emphasize defense-in-depth for accident prevention with assurance of passive safety response, and for accident consequence mitigation with characterization of phenomena. Within the framework of the SOPMB, shared activities include US testing experience, documentation of design measures for accident prevention and consequence mitigation, passive safety performance assessment in conceptual designs, and developmental methods for uncertainty quantification.

Recently, analyses of passive safety performance in oxide and metallic-fueled reactor concepts for 1 000 MWt and 2 000 MWt core sizes have indicated the potential for prevention of accident progression in ATWS. The modeling employed in these analyses is based on reactivity feedback and decay heat removal mechanisms demonstrated in EBR-II SHRT conducted previously.[6] The analysis results show that significant margins to reactor upset and damage (coolant boiling, cladding failures, fuel melting) can be assured by selection of appropriate design features, promising enhanced safety performance and improved economics.

In the Korea, a large scale sodium thermal-hydraulic test facility sponsored by the Department of Education, Science and Technology has been designed for verification of the design concept of the PDRC in a medium or large size pool-type SFR, focusing on assessing its cooling capability during the long and short term periods after reactor trip. Starting with the basic design of the test facility in 2008, its installation is scheduled to be completed by the end of 2011. The main experiments will commence in 2013 after the startup test in 2012.

The main test section of the experimental facility is composed of a primary heat transport system and a passive decay heat removal circuit which are scaled-down from the target design. The test section includes all major components in the primary heat transport system reflecting the real configuration. The preliminary concept of the main test section is shown in Figure 3.

Auxiliary fluid systems such as an intermediate heat exchanger gas cooling system, a sodium supply/purification system, a heat loss compensation system, and a gas supply system are included in the experimental facility. In order to represent important thermal-hydraulic phenomena in the PDRC as well as the reactor system, the main test section has been designed complying with proper scaling criteria for geometric, hydrodynamic and thermal similarities. Overall scaling of the facility is 1/125 for volume and 1/5 for height. The reactor vessel height and diameter are about 3.6 m and 2.3 m, respectively. The reactor core is simulated by electrical heaters of 1.9 MW capacity which corresponds to a 7% of the scaled full power. Sodium is used as a working fluid and its inventory of the main test section is approximately 13 tons. Operating temperatures of the reactor system are preserved in the experiment.

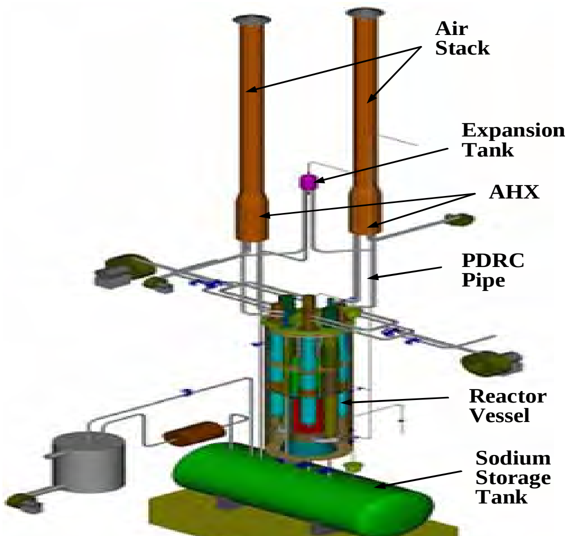

Figure 3: The preliminary concept of the PDRC test section

In the test, the natural circulation cool-down capability of the PDRC in conjunction with the reactor system will be investigated for various design basis events.

In the EAGLE-1 program,[5] several in-pile and out-of-pile tests were conducted under a co-operation between JAEA and National Nuclear Center of Republic of Kazakhstan. One of the main objectives of these tests was demonstration of effectiveness of the specific design concept to eliminate the severe re-criticality events in the course of core disruption accidents. Another important objective was acquisition of basic information on early-phase relocation of molten-core materials toward cool regions surrounding the core, which would be applicable to various core design concepts.

Figure 4 shows schematic of a typical in-pile test apparatus of the EAGLE-1 program. The geometry of this test apparatus is corresponding to a design concept equipped with a "discharge duct" within each fuel sub-assembly. The discharge duct of 2mm-thick stainless steel filled with liquid sodium was placed at the central part, and was surrounded by 75 UO2-fuel (BN350-type) pins with 400mm fissile height giving total fuel amount of ~8 kg. The test ID1 (Integral Demonstration test 1) was conducted with this test apparatus in IGR (Impulse Graphite Reactor). It was intended to produce a molten fuel-steel-mixture pool with the trapezoidal power diagram simulating the hottest part of the degraded core in an Unprotected Loss of Flow accident. This result showed a significant potential of core-material relocation even under a relatively low pressure difference (up to ~0.12MPa). These experimental data strongly suggested early fuel discharge with the inner duct equipped fuel subassembly design thereby eliminating large molten-pool formation which was the entry condition for severe re-criticality events. JAEA is presently conducting the EAGLE-2 program focusing on the long-term behavior of the degraded core with its stress on its coolability.

After 35 years operation, the 250 MWe (140 MWe since 1993) sodium cooled fast reactor Phenix was shut down on March 6th 2009. Before the decommissioning, the end of life tests are carried out during 2009 to collect relevant results in the fields of core physics, thermal-hydraulics fuel behavior under accidental conditions and negative reactivity transients occurred in '89-90'. The tests in the core physics field are: individual subassemblies reactivity worth, sodium void effect, control rod worth measurements by different methods on a low reactivity core and decay heat measurements. The thermal-hydraulics tests cover natural convection regimes in primary and secondary circuits and asymmetrical regimes in the primary vessel. One test concerns partial fuel melt in some experimental fuel pins.

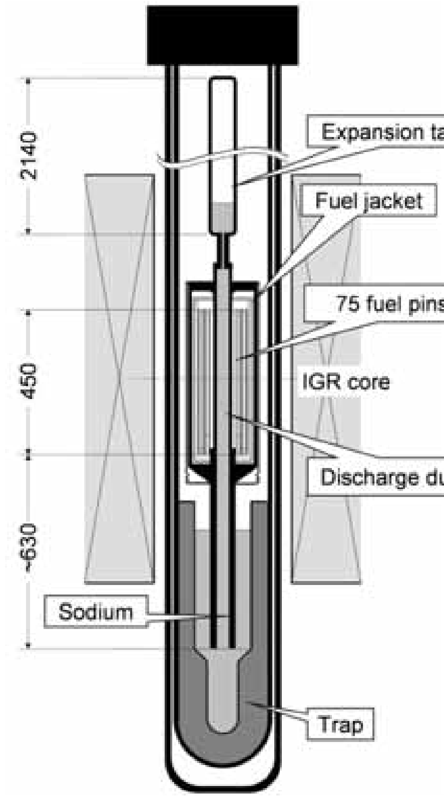

Figure 4: Typical EAGLE In-Pile Test Apparatus

Two tests are performed to help the comprehension of negative reactivity transients during the normal operation of the reactor, the first one concerns the neutronics and thermal-hydraulics coupling of an experimental moderated carrier with several adjacent blankets and the second is an artificially provoked core flowering to measure the reactivity effect. Two of these tests, thermal-hydraulic asymmetrical regimes ad core flowering are included in the scope of the SFR Safety & Operation Project. Main objectives are respectively, the validation of CATHARE ML system code and the understanding of safety related issues of the core neutronics/thermal-hydraulics/mechanics couplings.

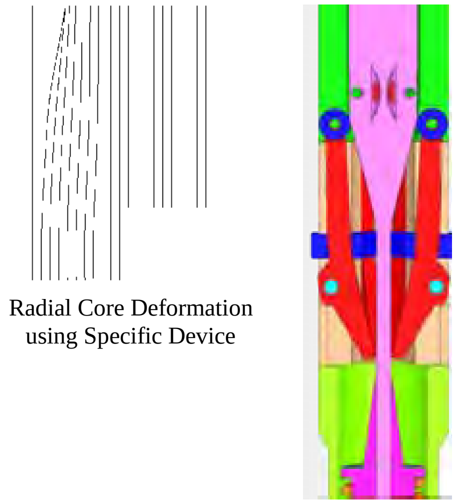

Figure 5: Phenix Core Flowering Test

VI. CONCLUSION

International collaborative R&Ds in the field of Safety and Operation Project are being implemented in the framework of GIF. The R&Ds include 1) analyses and experiments that support approaches and assess performance of specific safety features, 2) development and verification of computational tools and validation of models employed in safety assessment and facility licensing and 3) acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants. Implementing these R&Ds with other project R&D results will effectively achieve the development of Generation IV SFR.

References

1. *A Technology Roadmap for Generation IV Nuclear Energy Systems*, GIF-002-00, USDOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, Dec. (2002).

2. Kotake, S., *et al.*, *Development of Advanced Loop-Type Fast Reactor in Japan (1) Current Status of JSFR Development*, Paper 8226, ICAPP '08, Anaheim, CA, USA (2008).

3. Hahn, D., *et al., Design Concept of KALIMER-600, Global2005, Tsukuba, Japan (2005).*
4. Chang, Y., *et al*., *A Case for Small Modular Fast Reactor,* Global2005, Tsukuba, Japan (2005).
5. Konishi, K., *et al*., The result of a Wall Failure In-Pile Experiment Under the EAGLE Project, Nucl. Eng. Des., Vol.237 (22), p.2165, Nov. (2007).
6. Fistedis, S.H. (ed), The Experimental Breeder Reactor-II Inherent Safety Demonstration, Nucl. Eng. Des., Vol. 101, No. 1, (1987).

INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES (INPRO) AND ITS POTENTIAL SYNERGY WITH GIF

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Abstract – The IAEA's project INPRO was established in 2001 by bringing together technology holders, users and potential users to consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles. After completing development of evaluation methodology of innovative nuclear system in the area of Economics, Environment, Fuel Cycle and Waste, Safety, Proliferation Resistance and Infrastructure, the project moved to phase II that has four areas: Methodology development and its use by members, Future nuclear energy vision and scenario, Innovative technologies, and Innovation in institutional arrangement. Ten Collaborative Projects were started to address technical issues. The complementary relationship between INPRO and GIF has been defined by both groups and joint action plan was defined in 2008 April. Further areas of cooperation to create synergetic effect by utilizing unique added value of INPRO is considered and proposed in this paper.

I. INTRODUCTION

The IAEA has programmatic activities to stimulate technology development in order to assure the benefit from the use of NE for sustainable development by the use of innovative nuclear systems by paying attention to the needs of users and developing countries.

The IAEA's project INPRO is mostly funded by extra-budgetary contribution of its members. It was initiated in 2001 in order to provide a forum for discussion of experts and policy makers on all aspects of nuclear energy planning as well as on the development and deployment of innovative nuclear energy systems (INS). It brings together technology holders, users and potential users to consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles, but INPRO pays particular attention to the needs of developing countries. Currently there are 30 INPRO members (Figure 1) including five countries, which have not yet experienced operation of commercial nuclear reactors.

Figure 1: Current members of INPRO

II. STATUS OF THE PROJECT

The initial phase (2001-2006 summer) of INPRO has defined basic principles, user requirements and criteria in the area of Economics, Environment, Fuel Cycle and Waste, Safety, Proliferation Resistance and Infra–structure. After establishing a methodology usable by Member States in their evaluation and selection of INS, INPRO moved to the new phase (Phase 2) in the summer of 2006, which includes collaborative projects on technological issues that need to be addressed for improved economics, safety, proliferation-resistance and other topics. The current tasks in programme 2008-9 include the following and its progress as of the end of 2008 is reported in INPRO progress report: [1]

Task 1: INPRO Methodology.

Task 2: Application of Methodology by Members.

Task 3: Vision and scenarios on the use of INS for sustainable development

Task 4: Infrastructure needs and support framework for INS development and deployment.

Task 5: Common User Considerations by Developing Countries.

Task 6: Collaborative Projects.

Task 7: Communication & publications.

Since the early part of 2009, it was determined to streamline the project's task into the following four areas with a forum for dialogue by members as a cross-cutting vehicle for communication:

- Methodology development and its use by members.
- Future nuclear energy vision and scenario, Innovative technologies.
- Innovation in institutional arrangement.

II.A. Assessment methodology and its use

INPRO methodology published as TECDOC1434 [2] is consisting of a set of *Basic Principles*, *User Requirements*, and *Criteria* in a hierarchical manner as a basis for the assessment of INS in the areas of economics, safety, environment, waste management, proliferation resistance, physical protection and infrastructure.

Associated User Manual [3] has been made available to users in 2009.

Seven assessments of INS using this methodology have been completed by the end of 2008 and will be published soon as working material:

- Joint assessment based on a closed fuel cycle with fast reactors (Canada, China, India, Japan, Republic of Korea, Russia, Ukraine).
- Assessment of INS based on high temperature reactors (India).
- Assessment of additional nuclear generation capacity in the country for the period 2010-2025 (Argentina).
- Assessment of INS options for a country with small energy demands (Armenia).
- Assessment of the DUPIC fuel cycle with respect to proliferation resistance (Republic of Korea).
- Two independent assessment studies on IRIS and FBNR reactors (Brazil).
- Assessment of national INS (Ukraine).

These assessments also contributed to identifying the needs for R&D and also to provide valuable feedback for further improvements to INPRO methodology.

II.B. Common User Considerations by Developing Countries

IAEA General Conference Resolution in 2006 [GC(50)/RES/13B] required Agency to sets of common characteristics needed and desired by potential users of new nuclear power plants (NPP) in developing countries. In response to this, the INPRO started study by dialogue with experts in 54 developing countries, which included visit to the countries and workshops. The resultant expectations are reported as a NE Series document [4] and are characterized by:

- Competitiveness with alternative supported by comprehensive and reliable cost information.
- Suppliers role in financing.

- Supplier's role in establishing international mechanism for AOS of FC services, spare part pool.
- Proven by operation, standardized and licensed in the country of origin.
- Plant size distributed with 1GWe peak (Figure 2).
- Technology transfer, transfer of database of operational experiences of similar plants, local participation (some targets), support to (soft) infrastructure building.

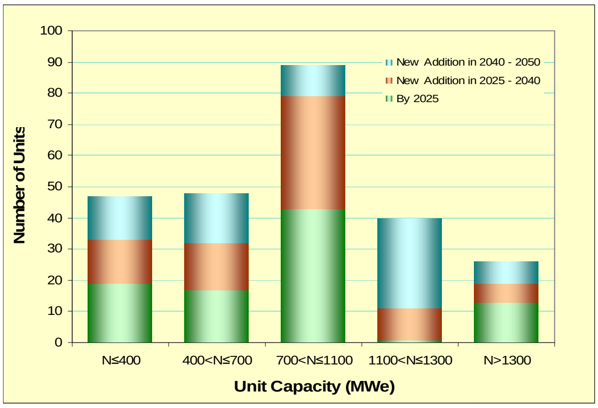

Figure 2: Plant size distribution

[Note] This distribution is based on the collective expectation by 31 experts in prospective user countries. It must be noted that assuming a rule of thumb (10% of grid, no interconnection with neighbors), among 54 prospective user countries, 20 countries have limitation to less than 300 MWe, and 12 countries to less than 700 MWe (larger than 300 MWe) as of today.

II.C. Collaborative Projects (CP)

INPRO members are identifying the needs of international collaborative projects or studies on a variety of topics that may be of common interest for countries expecting an increased role to be played by INS in the future. Three options are available to execute such projects;

- Coordinated Research Projects (CRP)
- Technical Cooperation projects (TCP)
- Joint Initiative (INPRO members establish a group and ask the IAEA for coordination)

Currently 12 CP have been planned and in various stage of implementation as is shown below on thematic basis. 10 of them are active by now.

- System analysis and fuel cycle: Thorium FC, GAINS (Global scenario analysis), RMI (Study of nuclear power programme under raw material limitation), FINITE (Fuel Cycle for INS), SMALL (issues of nuclear power and Fuel Cycle in small countries).
- Proliferation-resistance: PRADA (Acquisition/Diversion pathway analysis).
- Technology overview: AWR (Advanced water reactor thermal hydraulics), DHR (Benchmarking of code for analysis of decay heat removal from LMR), COOL (Coolant property and issues at elevated temperature).
- Safety: HTR H2 (Safety issues for advanced HTR for production of H2), PGAP (reliability of passive safety system), ENV (Benchmarking of environmental impact assessment code).

GAINS (Global scenario analysis) may deserve further elaboration here. GAINS assumes higher scenario, nuclear capacities reach 1 500 GW(e) by the mid-century and 5 000 GW(e) by 2100 and low scenario – 1 000 GW(e) and 2 500 GW(e), respectively. GAINS also assumes nuclear deployment scenario by three models; a 'homogeneous' model assuming that the whole world moves technically as one homogeneous group and "heterogeneous" with separate or synergistic cases.

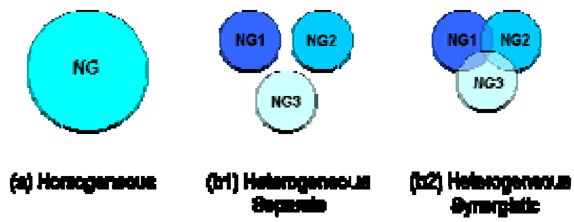

Figure 3: GAINS deployment models

Under GAINS, the non-geographical groups are defined as follows:

NG1: Countries which are most involved in the development and deployment of the INS and, consequently, would be able to incorporate them as soon as commercially available.

NG2: Countries with significant experience in the use of nuclear energy and most of the necessary infrastructure, but which are not quite ready to incorporate the most advanced nuclear energy system.

NG3: Countries supposed to incorporate nuclear energy in their energy mix, as newcomers.

A representative set of reactor types and fuel cycle installations and their expected time for introduction were also assumed. The current reactor fleet was assumed to be replaced gradually by new reactors such as large advanced thermal reactors (TR) and fast reactors (FR), small and medium reactors, HTR, ADS, and molten salt reactors. Different fuels (UOX, MOX, high density fuels, etc.) and different fuel cycles (U, Th) are considered. A very preliminary result indicates that, from the comparison of two models, an intensive cooperation in fuel cycle (heterogeneous synergetic model) enables twice high growth of the MOX fuel FR fleet globally as compared with the case of completely heterogeneous development.

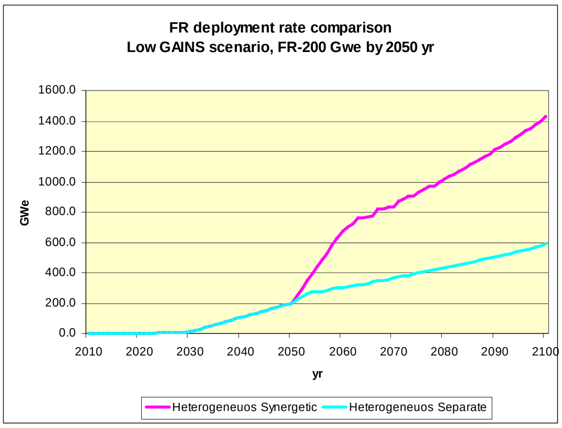

Figure 4: FR deployment rate comparison

In the 2010-11 programme and budget of the IAEA, two CRPs (Coordinated Research Projects) are being proposed: a) Simulation and modeling for development of technologies and b) innovative institutional approaches.

II.D. Publications

The results of INPRO activity is shared as public domain publications from the Agency.

INPRO has already published several documents with focus on methodology, namely 2008 Progress report [1], Assessment methodology (TECDOC-1434) [2] and accompanying manuals, [3] and Common User Considerations stage 1 report. [4] Further there are many soon-to-be published documents in 2009:

- Members' assessment results as working material (*Argentina, Armenia, Brazil, India, RoK, Ukraine, Joint Study)*.
- IAEA methodologies and tools for exploring long term energy development (including INPRO methodology).
- Lessons learned from Member's assessment.
- Legal and institutional aspect of non-stationary reactors.

III. FUTURE PERSPECTIVE

The generic direction of INPRO phase II activities is at this moment set for four areas as mentioned in section II.

One important point for discussion is that INPRO has initiated Vision/Scenario study intended for capacity building of MSs for their own long-term plan and to provide Member States with reference scenarios for consideration of maximizing the benefit from the use of NE for sustainable development. This activity is strongly linked with some CPs for global future nuclear energy system analysis such as GAINS and FINITE. If nuclear energy is deployed on a large scale in this century, the world would eventually have to consider recycling of spent fuel for better use of resources and other reasons.

INPRO will be able to provide a reference scenario and an opportunity for considering institutional and infrastructure issues jointly by technology holders and uses, to enabling conditions for the use innovative nuclear system using recycle of spent fuel.

IV. COLLABORATION WITH GIF

Relationship with GIF is a very often asked question. The web sites on each side have comprehensive information on this. [5][6] In essence, GIF is an international development activity by technology holders, whereas INPRO's has unique value as:

- A forum by both technology holders & users including countries not yet operating nuclear power plants.
- Addressing issues other than development.
- Having viewpoint from users.
- Paying attention to the needs of developing countries.

Their complementary relationship has been recognized in various occasions including G8 Summit in St. Petersburg in July 2006. [7]

Because the IAEA has a unique role (by statute) in safety and safeguard, the IAEA has been contributing GIF by sending experts to GIF working groups. INPRO has been participating in GIF Policy Group as observer. Occasionally interface meetings have been held and joint action plan has been established to create synergy by working together in such areas as use of IAEA Safety Standards for preliminary assessments of GIF systems, use of the GIF economic model ECONS by IAEA GCR group for cost estimates of GCRs, providing IAEA's HEEP code for non-electric application to GIF.

Future synergy could be developed, subject to discussion by both sides in interface meeting and INPRO Steering Committee (All the GIF members are members of INPRO. Through the national delegations to INPRO, GIF can express its expectations on synergy with GIF in the INPRO Steering Committee). In the author's view, this synergy could be created by utilizing unique value and activities by the IAEA and INPRO in the following areas, but not limited to:

1) Enhancing interface between technology users and holders of innovative nuclear energy system, IAEA/INPRO bringing users'/Developing Countries' point of view, and GIF bringing potentially available innovative technologies. This may include an assessment of selected GIF system using INPRO methodology from user's point of view.

2) Enhancing interface in the areas of safety, security and safeguard for establishing technologies to meet expectation for innovative systems, which could include assessments of GIF systems against IAEA Safety Standards.

3) Joint discussion on future reference deployment scenario of Generation IV systems.

4) Subsequent joint consideration of institutional and infrastructure conditions to enable expanded use of GIF systems including closed fuel cycle.

Further cooperation with INPRO and IAEA could benefit GIF when it considers the use of Generation IV systems in countries not yet operating nuclear power as of today. Given the situation that currently more than 60 countries are considering embarking on nuclear power programme, [8] GIF may consider what is the role of GIF in it and if GIF needs to re-orient its direction to meet the needs of all including the newcomers. Although the priority of the newcomers will be, as can be observed from the CUC document, [4] to prepare nuclear infrastructure and install proven reactors, it may be worth to consider in the course of development of Generation IV systems:

- Use of Generation IV systems by the newcomers in years to come in safe, reliable, secure and proliferation-resistant manner.
- Conditions to enable expanded use of Generation IV systems by countries

including newcomers (This consideration includes infrastructure for the use of closed fuel cycle and establishing waste repository and what institutional systems may enable this expansion.).

- SMR (Small and Medium size Reactor) version of Generation IV systems.

INPRO has been working and further intends to enhance its cooperation with the emerging countries through their application of INPRO methodology to evaluate INS and through CUC. [4] IAEA has been facilitating network of SMR development through its CRP (Coordinated Research Programme) and various Technical Meetings on SMR designs.

V. CONCLUSION

INPRO has evolved to include activities other than methodology development. They are Collaborative Projects, Common User Considerations, vision/scenario analysis, and consideration of institutional arrangement necessary to enable global use of innovative nuclear energy system. Synergetic effect by the INPRO working together with GIF will be possible based on the unique value of INPRO and complementary relationship.

References

1. International Atomic Energy Agency (IAEA), INPRO 2008 Progress Report, IAEA, 2009.

2. International Atomic Energy Agency, “Methodology for the assessment of innovative nuclear reactors and fuel cycles”: Report of the Phase 1B of the International Project on Innovative Nuclear Reactors and Fuel cycle (INPRO), IAEA-TECDOC-1434, IAEA, Vienna (2004).

3. International Atomic Energy Agency, “Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems”, IAEA-TECDOC-1575, IAEA, Vienna (Rev.1, 2008).

4. International Atomic Energy Agency, “Common User Considerations stage 1 report”, IAEA-NE Series Nr. NP-T-2.1, IAEA, Vienna (2009).

5. www.gen-4.org/GIF/About/faq/faq-interaction1.htm#3.

6. www.iaea.org/OurWork/ST/NE/NENP/NPTDS/Projects/INPRO/.

7. “Global Energy Security” statement, G8 Summit, (July2006).

8. Omoto, “IAEA activities in support of countries considering embarking on nuclear power programme”, NATO workshop, 2009 May 26-29, Armenia.

SESSION III SUMMARY / DISCUSSION

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SFR R&D activities have reached a high level of advancement within the GIF, making the SFR the most advanced system among those studied by the Forum members. Russia should soon be joining the US, Japan, France, Korea, China, EURATOM already cooperating on that system.

Its major activities can be summarized as follows. The System Integration and Assessment (SIA) Project Plan will be finalized in 2009 by the System Steering Committee (SSC) and the provisional SIA Project Management Board (PMB). As far as the Advanced Fuel project, various fuel irradiation tests in the PHENIX, ATR and Joyo reactors, were analyzed and evaluated. Within the Component and BOP project, external ultrasonic sensors and under-sodium visualization are under development, and a system to implement super-critical CO_2 Brayton cycle has been investigated. In the frame of the GACID (Global Actinide Cycle International Demons–tration) project, the analysis and evaluation of MA fuel material property and irradiated-fuel data has progressed, and the preparation for MA fuel material property measurement has started. Under Safety and Operation project, analyses and experiments are being carried out that support safety approaches and validate specific safety features.

GIF Policy Group has prepared a set of priority objectives for SFR for the next 5 years (see Appendix) as follows. In the field of Advanced fuels, priorities are 1) assessment of fabrication feasibility and irradiation behaviour of minor-actinide bearing fuels, 2) feasibility issues regarding actinide recycling and 3) preliminary selection of advanced fuels. As for the Safety approach, priorities have been indicated: 1) converging safety approaches including the case of severe accident, 2) to compare approaches and seek for consensus in the field of re-criticality and potentially positive reactivity coefficient issues, and 3) to provide solutions to the in service inspection issue. Challenges still remain in the fields of PHENIX, Monju and possibly CEFR and BN-800 test analysis and inter-comparisons, in the development of energy conversion systems and in the development of advanced materials, codes and standards.

The SFR Trilateral Collaboration among France, Japan and US, initially focused on reaching a common understanding of the mission and requirements for an SFR, has been found useful in order to support and share experimental demonstration facilities, with the potential to accelerate basic SFR technology development under the GIF framework.

Finally, the potential synergy of INPRO and GIF has been discussed, making use of unique features of the activities performed by IAEA and INPRO, and investigating the potential use of Generation IV systems in countries not yet operating nuclear power as of today.

There are still challenges to tackle with, in particular related to the implementation of the Fast Breeder Reactors paradigm (sustainability, resources extension, waste minimization): where and when there will be a potential market? Would fast reactors be able to help to establish a “low” or "zero" Carbon Society by 2050? Fast reactors came back from the state of lame duck under the "banner" of "MA burning" (like a true "Phoenix"…), and under a growing perception of their need for a global sustainability objective. Fast reactor role in order to ease waste disposal legacy and to expand the Uranium exhaustible resource, both are still justifiable goals to promote public commitment to SFR development and commercialization. Does this imply that alternatives should not been investigated? These are just a few questions that we need to face sincerely in order to communicate with decision makers and public. This is probably not only appropriate for SFR, but applicable to any so-called innovative nuclear system.

CLOSING SESSION

Chair: Yutaka Sagayama

CONCLUSIONS OF THE GIF SYMPOSIUM

Jacques BOUCHARD

Chairman: Generation IV International Forum

It is indeed a great honour for me to draw conclusive remarks from the Generation IV International Forum Symposium. This event, the first in the Forum's history, was organised at a time when the need for sustainable development of nuclear energy has become very acute worldwide. Its attendance turned out very high, with over 200 participants, inclusive of the GIF community members as well as special guests from each GIF member country.

The GIF has been a continuous, effective and very successful focal point for collaborative R&D activities for fourth generation nuclear systems, and the various presentations made during the Symposium highlighted accomplish–ments of the GIF work achieved so far.

On the various nuclear systems investigated by the GIF, some of the main results presented are as follows:

For liquid Metal reactors (Sodium cooled Fast Reactors – SFR-, Lead cooled Fast reactors – LFR-), international collaborative R&D activities are being successfully conducted.

In particular for the SFR,

- Candidate safety provisions & systems have been assessed and
- Preliminary evaluation of Minor Actinide bearing fuel, from irradiation tests performed in Phenix, ATR and Joyo has been performed.

For the LFR, a draft System Research Plan describes a dual track viability research program for both a small and a large system, with different missions.

For high Temperature Reactors (Very High Temperature Reactors – VHTR-, Gas-cooled fast Reactors – GFR-), there have been clear benefits of multinational collaboration in the GIF:

- Accelerating R&D for GFR & VHTR beyond needs of related near term projects
- Spurring the interest of process heat using industries in varied energy products of High Temperature Reactors

As for the other innovative systems (Super Critical Water Reactors – SCWR-, Molten Salt Reactors – MSR-):

- In the case of SCWR, the identification of two key areas, *i.e.* suitable materials and coolant chemistry have triggered the launching of two major collaborative R&D projects on these topics.
- In the case of MSR, reference configurations have been defined, allowing concentrating R&D on critical areas (liquid salt properties of reference compositions, qualification of high performance materials).

In summary, though much more work is needed to overcome some major technological obstacles, great progress has been made within the GIF. However, because the potential prospects of the various Generation IV systems are not yet fully established, it would be premature to eliminate any of the six technologies: *i.e*. no down selection should be performed at this point.

Finally, it appeared, from discussions on the topic "Towards industrial implementation: public and private initiatives interconnections" that:

- Government bodies stress that R&D shouldn't be performed without operators' views. Proper involvement of utilities and vendors even from the conceptual design stage is required.

- Industry stresses that any new plant's safety case should be convincing to Nuclear Regulators and the Public, with great care given to helping regulatory staff move from existing practices to those appropriate for new circumstances. Also, "real decisions" related to new concepts will be made largely on economical grounds

In conclusion, it is important to stress that the road to be followed before the Generation IV designs are attractive enough to allow for commercial deployment, is still long and paved with numerous hurdles. However, the preliminary results achieved by the GIF, and presented during this Symposium, clearly demonstrate that only joint collaborative efforts can ensure success. The Priority Objectives for the next five years, drafted out on the basis of all the results achieved so far by the GIF, show the path to follow.

APPENDICES

1. GIF priority objectives for the next 5 years
2. List of registered participants to GIF Symposium

Appendix 1

GIF PRIORITY OBJECTIVES FOR THE NEXT FIVE YEARS

GIF Priority Objectives for the Next Five Years

The GIF Symposium has the objective to give a global view on ongoing activities within the initiative. At the same time, the "Outlook" document illustrates the foreseen path forward. The following text provides a summary of agreed priority objectives for the different systems in order to help focusing and streamlining the GIF R&D activities during the next five years, consistent with GIF objectives. These priority objectives result from an analysis based on the following steps:

1) Review of the potential of the system.
2) Development target for the effective use of its potential.
3) Review of the current stage of development and analysis of technology options, with a view to down selection.
4) Assessment of key R&D issues and priority requirements.

These steps are discussed in the "Outlook" document. The summary presented below is essentially related to step 4) and provides for each system some key R&D priorities.

Very High Temperature Reactor (VHTR)

The VHTR has a long-term vision for operating with core-outlet temperatures in excess of 900 ºC and a long-term goal of achieving an outlet temperature of 1000 ºC. At the same time, the VHTR benefits from a large number of national programs that are aimed at nearer-term development and construction of prototype gas-cooled reactors that have adopted core-outlet temperatures in the range of 750 ºC to 850 ºC. The overall plan for the VHTR within Generation IV is to complete its viability phase by 2010, and to be well underway with the optimization of its design features and operating parameters within the next five years.

1. Core outlet temperatures

Objective:

- Further assess the range of candidate applications for VHTRs with the core outlet temperatures and unit power required, as well as the associated time line.

2. Domains of application and priorities

Objectives:

- Spur the interest of industries to use VHTRs to produce high temperature process heat in various industrial applications, thereby displacing fossil fuels and reducing the production of greenhouse gases.
- Make progress towards resolving feasibility issues (processes, technologies) and more reliably assessing performance;
- Update the definition of priority R&D needs.

3. Hydrogen production

Objectives:

- Make progress towards resolving feasibility issues (processes, technologies) and more reliably assessing performance of hydrogen production processes.
- Update the definition of priority R&D needs and pre-industrial demonstration projects.

4. Materials for the core and cooling systems

Objectives:

- Make progress towards resolving feasibility issues of high temperature design, including the qualification of heat resisting materials and manufacturing issues for key components of the core and the cooling systems (pressure vessel, intermediate heat exchangers).
- Update the definition of priority R&D needs.

5. TRISO fuel particles

Objective:

- Establish performance margins of the uranium-dioxide and uranium-oxicarbide coated particle fuels and establish fission product source terms.

Sodium Fast Reactor (SFR)

The SFR has a long term vision for highly sustainable reactors requiring its development in several important technical directions. At the same time, the SFR benefits from the worldwide operational experience of several sodium-cooled reactors and from a number of national programs aiming at nearer-term restart, development and construction of prototype Generation IV reactors. The overall plan for the SFR within Generation IV is to be well underway with the optimization of its design features and operating parameters within the next five years, and possibly to complete its performance phase by 2015.

1. Advanced fuels

In this area, after the identification of the advanced fuel options, major R&D efforts will be focused on fabrication feasibility and irradiation behavior of minor-actinide bearing fuels. A preliminary selection of advanced fuel(s) should be made.

The assessment of the high burn-up capability of advanced fuel(s) and materials should follow.

Objectives:

- Make preliminary selection of advanced fuels.
- Define priority irradiations beyond the Global Actinide Cycle International (GACID) project.
- Progress towards the resolution of feasibility issues regarding actinide recycling.
- Verify that milestones of the GACID project are realistic.

2. Safety approach

Objectives:

- Progress towards converging safety approaches.
- Revisit re-criticality and potentially positive reactivity coefficient issues, to compare approaches and seek for consensus.
- Assess, among other approaches, the effectiveness of inner-duct structures to mitigate severe accidents while enhancing fuel discharges without the formation of large molten-fuel pool. This assessment may benefit from analyses and conclusions of the EAGLE (Experimental Acquisition of Generalized Logic to Eliminate Re-criticalities) experiment if they can be shared with the international community.

3. In-service inspection

Research and development of in-service inspection approaches is following three parallel paths each of which is highly innovative in its own right. Significant improvements or breakthroughs in the ability to perform in-service inspection of in-vessel sodium components may result from this ongoing work.

Objectives:

- Draw conclusions from related R&D work and set priorities for the future.
- Progress towards resolving in-service inspection and repair feasibility issues.

4. Phenix, Monju and possibly CEFR and BN–800 tests

Objective:

- Summarize lessons learned from planned experiments and start-up.

5. Energy conversion systems

In this field R&D activities cover development and demonstration of sodium-CO_2 Brayton cycle advanced energy conversion systems including: the development and performance testing of compact heat exchangers; development and testing of small-scale sodium-CO_2 turbo-machinery and a complete integrated cycle; sodium-CO_2 interaction testing; CO_2 oxidation and carburization tests; and the analysis of system behavior for SFRs incorporating the sodium-CO_2 Brayton cycle.

Objectives:

- Draw conclusions from related R&D work and define priority research for the future.
- Make progress towards resolving feasibility issues on alternative energy conversion systems with gas or supercritical CO_2.

6. Materials, codes and standards

Objective:

- Develop of codes and standards for high temperature application (for example RCC-MR published by AFCEN is available and has been used for construction of PFBR).

Super-Critical Water Reactor (SCWR)

The SCWR has a long-term vision for water reactors that requires significant development in a number of technical areas. At the same time, the SCWR benefits from the resurgence of interest worldwide in water reactors as well as an established technology for supercritical water power cycle equipment in the fossil power industry. The overall plan for the SCWR within Generation IV is to complete its viability phase research by about 2010 and to operate a prototype fueled-loop by around 2015, thereby preparing for construction of a prototype reactor sometime after 2020.

1. Feasibility of meeting GIF Goals

The SCWR builds on a strong technical foundation from two advanced technologies: advanced Gen III+ water-cooled reactors; and advanced supercritical fossil power plants. The work performed to date does not show any issues regarding the viability of merging these two well-known technologies. However, the feasibility of meeting GIF goals and the estimation of the extent to which GIF metrics can be improved require significant R&D.

Objectives:

- Improve knowledge base to enable optimized designs and accurate assessments against GIF goals.
- Continue R&D needed to design and build a prototype.
- Continue conceptual designs of the various SCWR versions, including fast and thermal neutron spectrum designs using pressure tube and pressure vessel technologies.

2. Critical-Path R&D

Two critical-path R&D projects have been identified and are currently underway: materials and chemistry; and thermo-hydraulic phenomena, safety, stability and methods development.

2.1 Materials and chemistry

Objectives:

- Test key materials for both in-core and out-core components.
- Investigate a reference water chemistry taking into consideration materials compatibility and radiolysis behavior.

2.2 Basic thermal-hydraulic phenomena, safety, stability and methods development

Objectives:

- Continue investigating key areas such as heat transfer, stability and critical flow at supercritical conditions.
- Understand better the different thermal-hydraulic behavior and large changes in properties around the critical point compared to water at lower temperatures and pressures although the design-basis accidents for the SCWR will have similarities with conventional water-cooled reactors.

In addition, non-critical-path R&D areas will continue for specific designs in the areas of advanced fuels and fuel cycles (e.g., using thorium in the pressure-tube design and development of the fast-core and mixed-core options for the pressure-vessel design), and hydrogen production.

Gas-cooled Fast Reactor (GFR)

The GFR has a long-term vision for highly sustainable reactors that requires significant development in a number of technical areas. Unlike the SFR, the GFR does not benefit from operational experience worldwide and will require more time to develop. However, the GFR may benefit from its similarities with the VHTR, such as the use of helium coolant and refractory materials to access high temperatures and provide process heat. The overall plan for the GFR within Generation IV is to be well underway with the viability research within the next few years and to be completed by 2012.

1. Fuel

Work in this field focuses on assessment of multilayer SiC clad carbide fuel pins.

Objectives:

- Identify and demonstrate suitable technologies for pin fuels (low-swelling mixed-carbide fuel, multilayer composite SiC cladding for fuel pins).
- Update irradiation experiments in BR2, and identify other priority R&D needs (e.g., fabrication and behavior at extreme temperature).

2. Experimental demonstration design

The ALLEGRO experimental prototype is an option within the "European Strategic Research Agenda".

Objectives:

- Update and improve the definition of the experimental prototype ALLEGRO intended to demonstrate GFR key principles and technologies and to offer muti-purpose services such as fast-neutron irradiations and high temperature heat supply.
- Document ALLEGRO so as to support a decision around 2012 of proceeding towards detailed design studies and implementation.

3. Safety

GFR conceptual studies and operating transient analyses are priority R&D areas.

Objectives:

- Demonstrate the safety in case of depressurization accident;
- Study the phenomenology of severe accidents in core with ceramic cladding and structures;
- Confirm GFR safety through further accidental-transient analyses, assessments of innovative design features, and documentation of severe accidents analyses. Especially:
  - assess the merits of a pre-stressed concrete primary pressure boundary; and
  - proceed with tests of GFR fuel samples in extreme-temperature conditions.
- Further update the definition of priority R&D needs.

Lead-cooled Fast Reactor (LFR)

The LFR features a fast-neutron spectrum and cooling by an inert liquid metal operating at atmospheric pressure and relatively high temperatures. The main missions include the production of electricity, process heat, and hydrogen, and actinide management aiming at long-term fuel sustainability. The LFR has development needs in the areas of fuels, material performance, and corrosion control. The overall plan for the LFR is to be well underway with the development of its materials, design features, and operating parameters within the next five years.

1. Heavy liquid metal technology (coolant, materials, components)

Work in this field focuses on progress towards resolving issues related to the feasibility of heavy liquid metal technologies.

Objectives:

- Select and validate candidate structural materials.
- Demonstrate of corrosion control (with surface treatment, oxygen control, etc.).

2. Experimental demonstrations

Whilst the SFR remains the reference technology, the LFR and the GFR are promising alternatives. The LFR has a rather limited operational experience but it has several similarities with the SFR (e.g. fuel cycle). It was thus agreed within GIF that it should benefit from the relevant outcomes of the R&D on the SFR. An experimental reactor with a capacity in the range of 50 to 100 MWth will be needed to gain experience feedback by 2020.

Objectives:

- Update and improve the definition of the experimental prototype LFR.
- Confirm its feasibility and document its merits for testing LFR technologies in support of a decision around 2012 to proceed towards detailed design studies and implementation.

Molten Salt Reactor (MSR)

The MSR has a long term vision for highly sustainable reactors that requires significant development in a number of technical areas. The overall plan for the MSR is to be underway with the development of its design features, processing systems and operating parameters within the next five years.

1. Focus

In the United States, a PB–AHTR (900 MWth) has been selected as the lead commercial-scale plant AHTR concept.

In Europe, since 2005, R&D on MSR is focused on fast spectrum concepts (MSFR) which have been recognized as long term alternatives to solid-fuelled fast neutron reactors with attractive features (very negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle…). MSFR designs are available for breeding and for minor actinide burning.

Objective:

- Advance cooperative R&D work to further resolve feasibility issues and assess the performance of the different types of MSRs that have been considered.

2. Materials and on-line chemistry

A wide range of problems lies ahead in the design of high temperature materials for molten salt reactors. The Ni–W–Cr system is promising. Its metallurgy and in-service properties need to be investigated in further details regarding irradiation resistance and industrialization.

Objectives:

- Progress towards resolving feasibility issues and update priority R&D needs about structural materials for MSRs and on-line or batch-wise spent salt treatment processes.
- Plan for associated experiments.

Appendix 2

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