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GIF SYMPOSIUM PROCEEDINGS

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FOREWORD

Harold McFarlane
GIF Technical Director

It is a pleasure to introduce this report on the second Generation IV International Forum Symposium, held in San Diego, California on 14-15 November 2012. This report also represents the annual report on GIF activities for calendar year 2012.

In the twelve-year history of the Forum, the only opportunities for the entire GIF community to gather in one place have been at the two symposia. The first GIF symposium was held in 2009 and featured spirited discussions on such cross-cutting issues as advanced materials. It introduced GIF activities to some 900 participants at the GLOBAL 2009 meeting, provided a status report and outlook for the six Generation IV systems, and launched new initiatives in education and advanced simulation. The 2012 symposium calibrated the progress on technical achievements, collaboration, software tools and communication. The GIF leadership also laid out the strategic planning process that will provide the roadmap for the next decade of GIF activities. A full day of open sessions allowed participants in the American Nuclear Society Winter Meeting to attend the technical presentations.

On the eve of the symposium, ANS President Michael Corradini convened a special session of the Winter Meeting for a retrospective of the GIF's first decade. A summary of the remarks of former chairmen Bill Magwood and Jacques Bouchard can be found in this report. Both former GIF chairmen received Presidential Citations for their leadership in establishing the Forum. Current Chairman Yutaka Sagayama and Vice-Chairman Christophe Behar also spoke at the special session, with the essence of their comments also captured within the main body of the report.

Progress reports on the six reactor systems were the central focus of the symposium and remain the focus of this document. More than 200 people attended the open session where the reports were presented, and participant feedback indicated that this session was the highlight of the ANS meeting as well as the symposium.

On a personal note, it was an honour for me as the GIF Technical Director to have a leading role in organising both GIF symposia. I am profoundly grateful to the dozens of individuals who contributed to making the 2012 symposium a success.

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GIF: A SUCCESS STORY OF INTERNATIONAL CO-OPERATION

Jacques Bouchard
Former GIF Chair

The Generation IV International Forum (GIF) offers a large R&D co-operation framework aimed at preparing the development of future nuclear energy systems. More than ten of the most experienced countries in the design, construction and operation of nuclear power plants, are sharing knowledge and working together on innovative programmes geared towards the improvement of high-temperature and fast neutron reactor concepts.

Over the last twelve years, since the creation of the forum, co-operation has expanded, with new members joining and new projects being initiated. While it is always difficult to measure how efficient such co-operation on long-term development programmes really is, GIF is recognised as the first successful attempt to organise a large, worldwide framework for R&D co-operation in a sensitive field, close to industrial applications.

Among the reasons which can explain the success of GIF, I would like to emphasise three specific points: the world energy context, the choice of practical organisation and the strong support of the OECD/NEA.

At the turn of the 21st century, the world energy context changed considerably within a few years. While at the end of the 90s, everyone was still convinced that oil and gas would remain inexpensive for a long time to come, reducing the need to develop other energy sources, a few years later, fossil fuel prices rose strongly. This led rapidly to renewed interest in other energy sources, mainly nuclear and renewable technologies which provide the added benefit of helping to reduce greenhouse gas emissions. The concept of Generation IV reactors was launched in the United States in 2000 and the forum was established one year later. This was actually before the strong increase in fossil fuel prices but, in terms of market evolution, energy supply concerns were already on the minds of decision-makers at the time. Energy prices continued to rise in the following years and this can explain in part the relatively forthcoming support given by the governments of GIF member countries to the initiative. Retrospectively, GIF was the right initiative at the right time.

Creating a new international body to oversee R&D co-operation on Generation IV energy systems was an option that represented a challenge and required many legal and diplomatic efforts. Nevertheless, it was the option chosen, taking into consideration the well-focused objective of the initiative, the expected duration of the co-operation (i.e. long-term) and the specific limitations, in particular as related to industrial and intellectual property rights, while avoiding an unnecessary stacking of constraints from each participating organisation. As the scope of the co-operation under GIF was to initiate and manage innovative R&D programmes and did not include the design or the construction of prototypes, another important objective was to facilitate the exchange and sharing of knowledge and results, with each member country financing and managing its share of the common programmes. Significant efforts and negotiations between legal teams were necessary to ensure that this objective could be realised, maintaining clear lines of responsibility for each member and avoiding lengthy discussions between countries on monetary and work valuations. Another important decision was to organise R&D projects "à la carte", with each member selecting the concepts and projects in which it has an interest. This makes for more efficient decision-making processes, since they only involve those who are actually participating. As a whole, the organisation of GIF is rather simple given the scope of the co-operation, and has been well accepted by the members.

Among the simplifications adopted from the start, it was decided to have a single, light structure at the top, supported by the country holding the chair. However, it also became clear that a permanent technical secretariat was necessary to ensure continuity across evolutions in membership and rotations in chairmanships. The OECD/NEA agreed to provide this technical secretariat service, including the organisation of meetings, the safe-keeping of documentation and deliverables, the provision of legal advice and the handling of formal administrative matters. The choice of an international organisation with extensive experience in technical co-operation has proven to be effective, and the strong support of the NEA has considerably facilitated the establishment of the forum as well as its current operation.

GIF is now twelve years old. Innovation in Generation IV systems will still require many years of R&D co-operative efforts. The organisation itself can certainly be improved – and there is an ongoing initiative to do so – but its success in the end will depend on the willingness of members to achieve the overarching goals that were set a decade ago when preparing the GIF Technology Roadmap.

GIF AFTER TEN YEARS

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ANS President Michael Corradini joined with the meeting's general chair, Per Peterson, to host the ANS President's Special Session, "Ten Years since the Generation IV Roadmap: Progress and Future Directions for New Reactor Technologies", marking the first 10 years of the Generation IV International Forum (GIF). GIF also held a two-day symposium at the meeting, providing an update of its activities and discussing strategic planning for the next decade.

Peterson introduced the opening speaker, NRC Commissioner William Magwood, who earlier in his career was the longest-serving head of the Department of Energy's civilian nuclear energy program, serving under two presidents and five energy secretaries between 1998 and 2005. He oversaw the restoration of the federal nuclear technology program and the creation of the Nuclear Power 2010 program. Magwood also pushed forward Generation IV technology in the United States and internationally, serving as GIF's first chair.

When invited to speak at this session, Magwood said, he wondered whether after 10 years the truth about the creation of GIF could be told. To put the story in perspective, he referred back to the early days of the U.S. Atomic Energy Commission, when commercial nuclear power and other nuclear technologies were developed. In those days, he said, technology was about the future, "looking over the horizon, thinking about what was possible." It was not about fixing problems, as it is today.

But the loss of public enthusiasm for the technology, starting in the 1970s, and the plummeting of energy prices in the 1990s with the coming of natural gas as a major source of energy, took its toll on the DOE's nuclear program. In particular, Magwood said, the budget of the nuclear energy office effectively "zeroed" in 1998. But in fact, he noted, there was not a policy change. "We had come to the end of the planning cycles," he said. "The projects we had in mind were not attractive to people in Congress. We had terminated some of our projects [and] did not have things to do to replace them." He said that he and others at the DOE, some of whom were in the room, turned their attention to finding a way forward.

Jacques Bouchard (left) and William D. Magwood, IV (right) accepting Presidential Citations from President Corradini.

It soon became clear that to move ahead, it was important to show that "we were making a leap forward, we were taking the next step." He created the reactor generations diagram, with the next step being Generation IV. Then, because "we didn't have much money," Magwood said, they reached out to other countries, inviting those with nuclear programs to come to Washington to discuss next-generation technologies. The meeting took place in January 2000, and it turned out that all the countries that attended were of a like mind.

In a relatively short time, all the pieces needed for what became GIF fell into place – the goals for a Gen IV reactor were set, the concepts to move ahead with were identified, and an organization was put in place. This, Magwood said, may have been the first time that the world came together to decide on a fission technology to develop together.

Creating the GIF charter and putting together the technology road map went surprisingly smoothly, Magwood said, but establishing the GIF Framework Agreement took some time, particularly as lawyers became involved. Eventually, however, the problems were resolved and an agreement reached. The last difficult decision was where to hold the Framework Agreement signing ceremony. Magwood explained that if the event were held at the DOE, it would have been toasted with Diet Coke, and so it was held at the French Embassy – with champagne.

The story was not quite finished, however, as Magwood then revealed some of the actions taken by him and his team. In particular, he said, they didn't tell anyone they were establishing an international project. By "anyone" he meant senior managers at the DOE, who, he said, would have stopped it "in a heartbeat" had they known. The State Department was informed, he said, but the DOE's own managers were not, a situation that would not be possible today. Another key element was having the same group of people working on it. Without that longevity, he said, it would not have reached a successful conclusion.

Magwood had another message for the audience. Nuclear projects take a long time to develop, he said, and the DOE and the national labs as set up today are not able to support the sort of research that is required, not just for nuclear but for any engineering technology. He said that the nuclear community should seriously consider what changes are needed to make the nuclear research and development infrastructure work more effectively in the future, and he challenged ANS to take a leadership role.

Jacques Bouchard, former head of nuclear energy in France's *Commissariat à l'énergie atomique et aux énergies alternatives* (CEA) and chairman of GIF from 2006 to 2009, gave his perspective on GIF's creation, emphasizing that it has been a collective story. Many people put in a lot of effort to make it a success, he said, and should be highly satisfied with the results. From the European perspective, he said, the first surprise was that the United States came up with this cooperative initiative. The second surprise was to find a large consensus among the experts on most objectives, as well as on the technical aspects of the project. A third surprise was that the price of oil and gas began rising at just the right time.

Just before that happened, it was widely thought that oil and gas would remain relatively cheap for a long time, Bouchard said, while nuclear energy was being viewed as a temporary solution at best, and few thought that fast reactors had a future. It was a pessimistic time for the industry, he said, and the U.S. initiative was very much welcomed. At the same time, Russian President Vladimir Putin was also promoting international co-operation in the development of fast reactors.

Bouchard also noted the positive reactions from other countries. After a few meetings, the concept quickly evolved into an international project, which was formalized in the signing of the GIF charter in July 2001.

Among other achievements Bouchard mentioned was winnowing the initial 100-plus proposals down to the six basic Gen IV concepts in a relatively short time. This required consensus among many experts on a number of issues, such as an appropriate set of criteria that Gen IV systems should meet. The final selection of six concepts was made in July 2002, just one year after the charter was signed. The next step, developing the technical road map, was completed at the end of 2002.

At the time, the potential benefits of the advanced nuclear technology, such as more efficient use of resources and improved radioactive waste management, found support from many governments. Nuclear power also fit into the "hydrogen economy" concept that was being promoted at the time. While governments have other concerns today, GIF is now firmly established and working well, Bouchard said.

A decision that turned out particularly well for GIF was to have a "light" organization. Instead of creating a large administration and budget, he said, the program is effectively paid for by in-kind contributions, which has achieved very effective co-operation among the participants.

THE GENERATION IV INTERNATIONAL FORUM ADVANCEMENT AND OBJECTIVES

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ABSTRACT

In January 2000, under the development initiative of the Generation IV nuclear energy systems (Gen IV systems), the U.S. Department of Energy (DOE) held a workshop in Washington D.C., joined by the IAEA and the NEA as well as nine countries from Argentina, Brazil, Canada, France, the UK, Japan, Korea, South Africa and the United States. On that occasion, a joint statement was made declaring the promotion of technology development for the Gen IV systems to ensure energy security, and at the same time the Generation IV International Forum (GIF) was formed as an international community for the development of the Gen IV systems. In July 2001, the GIF Charter was signed and GIF was officially established. Subsequently, in 2002, six concepts, Sodium-cooled Fast Reactor (SFR), Very High Temperature Reactor (VHTR), Supercritical Water-cooled Reactor (SCWR), Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR), and Molten Salt Reactor (MSR), were selected as Gen IV systems, and the GIF Technology Roadmap was formulated. This paper introduces the current activities including collaborations with INPRO, the message to Fukushima accident, the safety design criteria for SFR, and the decadal strategic planning. Under the situation of needs for more nuclear energy by growing energy demands and global environmental challenges, the importance of the GIF activities is increasing.

I. INTRODUCTION

Demand for electric power is increasing in today's world and will continue steady growth. For that, capability of supply to keep up with the growing demand will be necessary. It is clear that nuclear energy is the greatest means to cover the rising power demand. Actually many countries are promoting nuclear energy as national projects. The necessity of nuclear technology and development will certainly become more important to sustain stable energy supply also in global perspective.

II. GENERATION IV NUCLEAR ENERGY SYSTEMS

Generations of nuclear energy systems are classified into four categories. Generation I, the prototype reactors that started operation in 1950s and the early 1960s, Generation II, the commercial reactors that had been constructed from the late 1960s to the early 1990s, and Generation III, the advanced models of the Generation II reactors that started or will start operation from the late 1990s to early 2010s. Currently most reactors in operation are Light Water Reactors (LWRs) of Generation II or, for the more recently constructed plants, Generation III design.

Generation IV (Gen IV) nuclear energy systems (Gen IV systems) are innovative reactors that will enable nuclear energy to meet the energy needs in the future while also complying with the concept of sustainable development, in particular relating to more efficient use of uranium and optimized management of nuclear waste. Gen IV systems will also have enhanced safety, competitiveness and proliferation resistance.

GIF is an international co-operation framework that was formed for the development of Gen IV systems and the members now consist of twelve countries and EU. Its purpose is to share and carry out necessary R&D among member countries.

GIF has four technology goals of safety and reliability, sustainability, proliferation resistance and economic competitiveness that are required for innovative nuclear reactor concepts.

R&D activities are divided into three phases for each Gen IV systems. R&D on viability and performance phases has been promoted for selected six Gen IV systems such as SFR, VHTR, GFR, SCWR, LFR and MSR.

For the SFR system, four project arrangements have been concluded. A project arrangement of System Integration & Assessment (SI&A) is ready for signing. Concluded four project arrangements are Advanced Fuel (AF), Global Actinide Cycle International Demonstration (GACID), Component design and Balance-of-plant (CDBOP), and Safety and Operation (SO). In the AF project, the R&D have been conducted for selection of high burn-up Minor actinide (MA) bearing fuels, cladding and wrapper tubes withstanding high neutron doses and temperatures. GACID project sets out to demonstrate on a significant scale that fast neutron reactors can manage the whole actinide inventory, as Fast Reactors can transmute minor actinides (Neptunium/ Americium/Curium) and thereby reduce the risk of high level radioactive wastes and proliferation. In this project, Joyo and Monju are scheduled to be used. The objects of the CDBOP project are to enhance SFR system performance through development of advanced components aiming at economy improvement and R&D for supercritical CO_2 cycle as effective power conversion concepts. SO R&D purposes are to analyze and experiment supporting safety approaches, to develop computational tools and acquire reactor operation technology.

In VHTR system, three project arrangements were concluded to implement irradiation tests as to graphite & fuels and to research hydrogen production, etc. These projects are progressing steadily. Computational Methods Validation and Benchmarking (CMVB) project is in a preparatory stage. The hydrogen production (HP) project is to evaluate feasibility, efficiency and economics of hydrogen production for two main processes of the sulfur-iodine thermo-chemical cycle and the high-temperature electrolysis process. The Fuel and Fuel Cycle (FFC) project works to increase the understanding of standard design composing of UO_2 kernels fuel with SiC/PyC coating and examine the use of UCO kernels and ZrC coatings for enhanced burn-up capability, reduced fission product and increased resistance to core heat-up accidents above 1 600 C. The Materials project is aiming at the development and examination of design codes & standards for graphite, metals and ceramics.

The GFR system has two projects. Fuel and Core Materials (FCM) project is being prepared for official agreements, and Conceptual Design and Safety (CD&S) is in progress. The CD&S project is to define a conceptual design with operating parameters and appropriate safety architecture.

In the SCWR system, two projects are progressing for evaluation of system concepts, optimization of performance and selection of materials. One more project -fuel qualification test project- is currently negotiated.

Memoranda of Understandings (MOUs) have been concluded for the LFR and MSR systems. System research plans are being prepared for both systems.

III. RECENT GIF ACTIVITIES

In this paper, four representative examples of GIF activities are taken up. Firstly, synergy effects are expected through collaboration with the IAEA's International Project on Innovative Nuclear and Fuel Cycles (INPRO). Both GIF and INPRO are multilateral international cooperative frameworks for R&D of the next generation nuclear systems. So discussions and information exchanges are made for development of nuclear energy systems. Interface meetings have been held once a year, and SFR safety workshop (WS) has been conducted intensively. In the SFR safety WS, the GIF side has explained the progress of safety design criteria (SDC) to INPRO. Fields of co-operation other than safety, there are methodology development including economics and non proliferation.

Secondly, the GIF announced at Policy Group (PG) meeting last October that it was confirmed to continue co-operation among GIF members and to promote the development of the Gen IV systems, while taking in the lesson learned from the accident of the Fukushima Daiichi NPPs by the tsunami occurred on March 11, 2011.

Thirdly, the GIF has drafted SDC for SFR as international common standard. This was proposed in the South Africa PG meeting in October 2010. After that, a task force was made under the PG. In order to achieve enhanced safety for Gen IV systems, the reinforced defence-in-depth is incorporated, considering measures on prevention and mitigation of severe accidents. The lessons learned from Fukushima nuclear accident are also taken into account in the SDC. The draft of SDC will be reviewed at San Diego PG meeting.

Finally, next decadal strategic planning is under intensive review aiming at further progress of GIF. Considering ten years have passed from establishment of GIF, R&D of Gen IV systems has been developing into new phases, three task forces for technology roadmap update, strengthening R&D collaboration and strengthening ties with other international organizations were set up to draft strategic planning of each fields.

IV. SUMMARY

The GIF has played a major role as an international framework for development of next generation nuclear energy systems since 2001. With increasing energy demands and global environmental challenges, demands for nuclear energy is expected to further expand in the future. Especially, rapidly energy demand expanding countries including China and India plan to enlarge the capacity of nuclear power. So the Gen IV systems also need to be deployed. Even after unexpected Fukushima Daiichi NPPs accident, Gen IV systems' importance has not changed. More enhancement of safety is crucial. For further development of Gen IV systems, strengthen of R&D co-operation within GIF and fruitful co-operation with other organizations are required.

SAFETY ASSESSMENT FOR GENERATION IV NUCLEAR ENERGY SYSTEMS

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ABSTRACT

The Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) was created to promote a consistent approach on safety, risk, and regulatory issues for Generation IV systems. Early activities of the RSWG focused on identifying and defining the attributes and characteristics that might help to ensure the fulfilment of Generation IV safety goals. More recent work was directed at creating a coherent framework to assess the safety of Generation IV systems, and to contribute to the development of Generation IV concepts in which risk and safety insights derived from the assessment methodology are actively used throughout the design process.

This paper describes the GIF's Integrated Safety Assessment Methodology (ISAM), its development, and its applications to date. The development of the ISAM included interactions with, and input from, multiple stakeholders and nuclear safety experts. Based principally on probabilistic safety assessment, and offering assessment tools well suited to all stages of design development, the ISAM is intended to offer system developers a flexible and powerful safety assessment methodology comprised largely of currently accepted and validated tools. Limited scope trial applications of the methodology have been conducted and feedback has been consistently and strongly positive. Some challenges exist, however, and current work of the RSWG is aimed at enhancing the utility of the ISAM, and supporting its use within the six System Steering Committees. The nature of these challenges, and the RSWG's efforts to improve the value and usability of the ISAM are also discussed in this paper.

I. INTRODUCTION

The principal purpose of the Generation IV International Forum's Risk and Safety Working Group (RSWG) is to promote an effective and consistent approach in ensuring the safety of Generation IV nuclear systems. Early work of the RSWG focused on development of a coherent safety framework for Generation IV nuclear systems, and on identifying safety characteristics that could help to achieve Generation IV safety goals. The results of this work were published in a 2007 report entitled, "Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems". Following the publication of that report, the work of the RSWG turned to the development and definition of a safety assessment methodology to support the development of Generation IV systems. This paper discusses that methodology, its applications to date, and future work of the RSWG.

II. PURPOSE OF THE INTEGRATED SAFETY ASSESSMENT METHODOLOGY FOR GENERATION IV SYSTEMS

The Integrated Safety Assessment Methodology (ISAM) is intended to be used in three principal ways:

- **Influence the course of Generation IV system design**

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

- **Support risk and safety comparisons**

  The methodology can be applied at any point in the design process from the conceptual development phase through the final design phase to support risk and safety comparisons of various nuclear system concepts and designs. In this application within a design concept, the methodology can form an input to "down-select" and formulate decisions requiring a systematic and comparative understanding of safety issues predicated on a common analytical framework.

- **Qualitatively characterize and quantitatively measure the level of safety and risk**

  The ISAM can be applied throughout the design process to measure the level and quality of safety and risk associated with a given design relative to a specified safety objective or licensing criterion. In the late stages of design maturity the ISAM will allow evaluation of a particular Gen IV concept or design relative to various potentially applicable safety metrics or "figures of merit". This post facto application of the ISAM might 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 ISAM methodology be used to dictate design requirements, to dictate compliance with quantitative safety goals, or to constrain designers in any other way. The sole intent is to provide a methodical approach that contributes to the attainment of Gen IV safety objectives, that yields valuable insights into the nature of safety and risk of Gen IV systems, and that permits meaningful comparison of the safety of Gen IV concepts.

III. ISAM DESCRIPTION

As the RSWG set about defining a suitable safety assessment methodology to be used in the ways described above, the group first thought about the characteristics that such a methodology should exhibit. These attributes include the following:

- 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 lengthy validation.
- The methodology must allow for the integration of a diverse range of multidisciplinary inputs including those that are principally qualitative and those that are principally quantitative in nature.
- The methodology should offer flexibility allowing applicability to all stages of design development, and to technical safety issues of varying complexity.
- The methodology should reflect both probabilistic and deterministic perspectives, inputs, and outputs.
- 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 methodology must help identify areas that need additional research, data collection, and improved analytical models.

The ISAM provides an integrated set of tools that satisfies the list of desired attributes outlined above. It offers a Risk Informed approach in which qualitative and quantitative, deterministic and probabilistic insights are made available to support the designer throughout the design process.

The integrated methodology consists of five distinct analytical tools, or "elements". These include:

- Qualitative Safety Features Review (QSR).
- Phenomena Identification and Ranking Table (PIRT).
- Objective Provision Tree (OPT).
- Deterministic and Phenomenological Analyses (DPA).
- Probabilistic Safety Analysis (PSA)

It is intended that each element be used to answer specific safety-related questions with different 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 elements can be selected for individual and exclusive use, the full value of the integrated methodology is derived from the complementary use of all elements in an iterative fashion throughout the development cycle. For additional detail regarding the ISAM methodology, see Reference 1.

IV. ISAM RELEVANCE, TRIAL APPLICATIONS AND FEEDBACK

From the outset, the RSWG consulted with interested and knowledgeable stakeholders to ensure that the ISAM would be a highly useful, technically valid assessment tool. In addition to the fact that RSWG membership includes participation by a diverse group of international nuclear safety experts, the methodology has benefitted from inputs from, and interactions with, representatives of all Generation IV System Steering Committees, the Generation IV Senior Industry Advisory Panel, and the Generation IV Experts Group. In addition, the ISAM methodology has been shaped by inputs from representatives of the International Atomic Energy Agency IAEA), the Multinational Design Evaluation Program (MDEP), and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). Further feedback was received from members of the international nuclear safety community when the ISAM formed a major focus of the SARGEN IV workshop in March 2012. As a result of these collaborative interactions, the ISAM represents a strong consensus among stakeholders in terms of technical validity, expected value, and suitability for its intended purposes.

A workshop held in April 2010 at the Joint Research Center at Petten, the Netherlands, brought together representatives of each of the six Generation IV System Steering Committees (SSCs), along with the membership of the RSWG. The major purpose of this workshop was to present the ISAM in detail to the SSCs, and to solicit feedback on the expected value and practicality of the methodology.

Subsequent to that workshop, at least two limited scope trial applications of the ISAM have been completed – one for a Japanese Sodium Fast Reactor concept, and one for a French Sodium Fast Reactor concept. The major purpose of both trial applications was to demonstrate application of the ISAM to a realistic, developing advanced reactor development effort.

To date, feedback received from all stakeholders indicates a very strong consensus regarding the utility of the methodology. The only concerns or issues that have been noted in either the workshop or as a result of the trial applications relate to the level of expertise that is required to apply the integrated methodology. Partly as a result, it has also been suggested that additional guidance concerning application of the ISAM would be helpful. Based on that input from interested stakeholders, current activities of the RSWG are shifting toward the development of such guidance.

V. FUTURE RSWG WORK

With the framework and elements of the ISAM well defined, methodological work of the RSWG has turned to development of guidance for ISAM applications. The RSWG will continue efforts to strengthen and maintain interfaces with the SSCs, and will help support ISAM applications by those SSCs to the extent possible. In addition, the RSWG will continue its interfaces with INPRO, IAEA, MDEP, and others interested in advanced reactor safety.

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PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION WORKING GROUP: METHODOLOGY AND APPLICATIONS

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ABSTRACT

We summarize 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 designs as well as to developing approaches to advanced safeguards.

I. INTRODUCTION

After the Generation IV International Forum (GIF) Roadmap [1] was issued in 2002, the Proliferation Resistance and Physical Protection Working Group (PRPPWG) was established and charged with developing measures and metrics for expressing proliferation resistance and physical protection and an associated evaluation methodology. In the R&D program for PR&PP, it was envisioned that R&D would be conducted in three areas: (1) safeguards and physical protection technology R&D for each GIF system, (2) formulation of PR&PP criteria and metrics, and (3) evaluation of the criteria and metrics. The PRPPWG was established in late 2002 with a charter that covered items (2) and (3). Specifically, the Working Group was charged with developing a methodology for the systematic evaluation of proliferation resistance and physical protection of Generation IV energy systems. Overall, the method would enable comparative evaluation of the performance of different systems (or options for a given system) against the GIF PR&PP goal. The Working Group would also determine the measure (or measures) for expressing proliferation resistance and physical protection, and develop an evaluation approach that is comprehensive and quantitative to the extent possible.

The PRPPWG was not given a specific mandate with respect to item (1). As the 2002 Roadmap outlines, each GIF design would support R&D on material deployed, potential vulnerabilities, protective barriers, safeguards approaches, potential misuse, material protection, control and accounting for each step in the fuel cycle, etc. While each GIF design has not yet formally explicitly addressed all nine tasks given in the 2002 Roadmap [1] for PR&PP R&D, there has been interaction between each of the System Steering Committees (SSCs) and the PRPPWG on the status of each design with regard to PR&PP R&D and a joint report between the PRPPWG and the SSCs was approved by the GIF Policy Group in 2011 (see discussion below).

Since the issuing of the GIF Roadmap and the establishment of the PRPPWG, the importance of considering safeguards needs as early as possible in the technology design process (Safeguards by Design) has become widely recognized, as well as the importance of integrating the considerations of safeguards, security, and safety (the 3S approach – see Reference [2]). In this respect the interaction of the SSCs with the PRPPWG, the engagement of the individual design teams with the PR&PP process, and the dual consideration of security and safeguards concerns within the PR&PP process, demonstrates the alignment and leadership of GIF in the area of international PR&PP development over the last decade.

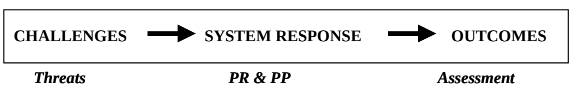
In parallel to the development of the methodology, the group has promoted the concept of safeguardability defined as the degree of ease with which a system can be effectively and efficiently placed under international safeguards [3].

II. DEVELOPMENT AND APPLICATIONS OF THE PR&PP METHODOLOGY WITHIN GIF

In a succession of revisions beginning in 2004, the PRPPWG has developed a methodology for PR&PP evaluation for all GIF systems. Measures and associated metrics were included in each revision. Consensus was achieved amongst all participating GIF countries and related organizations (IAEA, EU) and Revision 6 of the methodology report was approved by GIF for open distribution in 2011 [3].

Figure 1 illustrates the methodological approach at its most basic. For a given system, analysts define a set of challenges, analyze system response to these challenges, and assess outcomes.

Figure 1: Basic framework for the PR&PP evaluation methodology



The challenges to the nuclear energy system (NES) are the threats posed by potential proliferant States and by sub-national adversaries. The technical and institutional characteristics of the Generation IV systems are used to evaluate the response of the system and determine its resistance to proliferation threats and robustness against sabotage and terrorism threats. The outcomes of the system response are expressed in terms of PR&PP measures and assessed.

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 robustness 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.

The methodology was developed, demonstrated, and illustrated by use of a hypothetical "example sodium fast reactor" (ESFR), by members of the PRPPWG [4]. The ESFR case study was the first opportunity to exercise the full methodology on a complete system, and many insights were gained from the process. In particular, the approach of breaking the assessment into subtasks, each focusing on a separate area of PR&PP (diversion, misuse, breakout) handled by a dedicated subgroup with diverse international membership, was useful in generating new insights and concept development.

Workshops with GIF designers and other stakeholders, to familiarize them with the methodology and to understand their needs for the design process, were held in the USA, Italy, Japan, the Republic of Korea, and (most recently) Russia. This has helped to address one challenge with PR&PP, which is the engagement of designers since PR&PP has typically been a topic tackled in the latter stages of design and at the initiation of external bodies like the IAEA or Euratom. These workshops have spread awareness of the PR&PP methodology beyond the GIF community, which is appropriate since the methodology itself is applicable to the whole range of nuclear technology.

Starting in 2007, the PRPPWG and the six SSCs conducted a series of workshops on the PR&PP characteristics of their respective designs and identified areas in which R&D is needed to further include such characteristics and features in each design. A common template was developed to collect in a systematic way Gen IV design concepts, information, and PR&PP features and issues. This work culminated with (six) reports written jointly by the PRPPWG and the SSCs for each design. An overall report was approved by GIF for open distribution in 2011 [5]. The intent is to generate preliminary information about the PR&PP merits of each system and to recommend directions for optimizing their PR&PP performance.

The report captures the current salient features of the GIF system design concepts that impact their PR&PP performance. It identifies crosscutting studies to assess PR&PP design or operating features common to various GIF systems; and it suggests beneficial characteristics of the design of future nuclear energy systems, beyond the nuclear island and power conversion system, that should be addressed in subsequent GIF activities.

A summary of the work of the PRPPWG over the past decade appears in a special issue of the ANS journal *Nuclear Technology* in July 2012, Volume 179, Number 1 on the topic of safeguards. Several papers on the methodology and its applications, authored by members of the PRPPWG, appear in this issue.

III. APPLICATIONS OF THE PR&PP METHODOLOGY WITHIN NATIONAL PROGRAMS

Others, in national programs, have adapted the PR&PP methodology to their specific needs and interests:

- In the USA, the methodology has been used to evaluate alternative spent fuel separations technologies [6].
- In Canada there has been a safeguards-by-design use of the PR&PP methodology in the licensing process for two new CANDU designs [7].
- In Belgium the PR&PP methodology was used in the PR analysis of the MYRRHA accelerator-driven system [8].
- Elsewhere in the EU, the PR&PP methodology is also being applied for providing PR consideration within a European R&D project on a Sodium Fast Reactor [9].

IV. IAEA INTERACTION

The PRPPWG has coordinated closely with the IAEA since its inception; i.e., there has always been an IAEA representative on the PRPPWG who has contributed to the work and direction of the group.

In terms of methodology development there has been considerable interaction between GIF and the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) PR assessment methodology [10], beginning with a comparison of the respective methodologies of the two organizations with an aim towards understanding how prospective users could benefit from each and from a joint application of the approaches. Some members of GIF have participated in INPRO projects and other IAEA projects in nuclear energy and safeguards which has provided a useful catalyst to further co-operation. Moreover the regular annual meetings between GIF and INPRO have provided an excellent forum for information exchange and for defining future collaborative efforts.

Work that has recently been initiated under INPRO's PROSA (Proliferation Resistance and Safeguardability Assessment) project will be monitored for potential application in the GIF program. One of the goals of PROSA is to develop a workable assessment approach that will potentially draw upon the GIF PR&PP methodology to fill gaps in the INPRO approach, leading to a unified process.

There are several benefits that accrue from continued interaction between GIF and the IAEA, and there is a strong argument for the complementary nature of the two methodologies:

- The IAEA/INPRO methodology for non-proliferation provides "rules of good practice" for design concepts. It thus provides a useful checklist that ensures that technology assessors "did things right".
- The GIF/PRPP methodology is a systematic approach to evaluating vulnerabilities in design concepts. It thus provides the analysis approach that an INPRO assessment might utilize (as currently discussed in the PROSA project), and helps to make sure that assessors "did not do things wrong".

Together the methods could provide users with an overall approach to assuring robust future designs. IAEA/INPRO is more broadly known to IAEA community; GIF/PRPP provides a powerful analytical tool for evaluating strong and weak spots and therefore reducing proliferation risk in a design. Together, both products are potentially useful in national programs.

V. CURRENT SITUATION ASSESSMENT

Currently the PR&PP methodology is the most comprehensive publicly available evaluation methodology for any technology – despite being developed specifically to meet GIF goals. The PR&PP methodology is reasonably complete as an overarching framework; however, specificity of techniques and applications are needed, primarily as determined by the user.

With the interaction with designers, a need has emerged for simplified scoping PR&PP evaluations. Such scoping applications are a valid application of the methodology, and in fact support the view that PR&PP can be implemented at the earliest stages of design when a focused and simplified approach is appropriate. The application of the PR&PP methodology in Canada, noted above, was a pared-down implementation in this category.

Some observers are calling for a more simplified version of the PR&PP methodology to enable usage by newcomers. It is the view of the PRPPWG that, while it might be beneficial to create a high-level "guidance document" that lays out the steps to an evaluation and directs users to the relevant sections of the methodology, it is not advisable to simplify the methodology itself for generic application, since this carries a risk of omitting relevant components. However, each evaluation should define its scope and goals including a possible tailoring of the needed approaches.

In the international safeguards community, the concept of Safeguards by Design (SBD) has emerged as a key "cultural shift" to be encouraged amongst designers, and as noted earlier GIF was one of the first development organizations to embrace this concept through its creation of the cross-cutting PRPPWG. There are ongoing and planned efforts both in national programs and internationally, by IAEA and by the EC, to promote and implement SBD in the nuclear facility design process. IAEA has efforts underway on SBD [11] and is likely to publish a guidelines document in 2012 and facility-specific guidance documents are expected to be published in 2013-14. As noted above, IAEA also has the PROSA program underway which will have relevance to SBD and PR&PP.

There is an increased emphasis world-wide on the development and deployment of small modular reactors (SMRs). Since some of the GIF designs are in the SMR category it will be important to maintain cognizance of issues and developments as they pertain to PR&PP. While some SMRs share many characteristics of relevance to PR&PP with conventional reactors, others – particularly those with advanced fuel cycles or those destined for remote operation – represent novel designs or implementations that will benefit from a consistent and comprehensive PR&PP evaluation at various stages of the design process.

It will be important to maintain cognizance of post-Fukushima lessons-learned for their potential relevance to PR&PP.

A committee of the US National Academies is currently studying how methodologies for "proliferation risk assessment" relate to the needs and questions of policy makers in this area. Their findings and recommendation will be issued in March 2013.

VI. FUTURE PR&PP ACTIVITIES

Working with GIF SSCs on maturing their designs: As new and innovative design are developed for nuclear energy systems through GIF (and possibly others), 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.

If the GIF sponsors in the various participating countries wish to advance the utilization of PR&PP methods in the design process, the next major step for joint activity between the SSCs and the PRPPWG should be to designate one or two concept designs for an in-depth pilot study. This would involve applying the PR&PP methodology to the development of a model of the design and would be a follow-on effort to the initial joint studies between the PRPPWG and the SSCs that have been described above. The model would be rather high-level and attempt to capture the broad features of the design in terms of expressing its robustness for PR&PP characteristics. The pilot study would include participation by nuclear energy system designers as specified by the SSCs and members of the PRPPWG who would bring modeling expertise to the collaboration. In addition, subject matter experts in safeguards and physical protection would be needed to provide specific context for the development of the models.

This study could fit well within the scope of one of the Gen IV System Integration and Assessment (SIA) projects.

In the longer term, when the results and insights from these pilot studies become available, other GIF design concepts would also engage in such model development with the assistance of the PRPPWG. The overall benefit would be to introduce PR&PP early in the design process in order to cost-effectively provide for safeguards and security before the design has fully matured (and to thus avoid costly retrofits). This would ultimately be a useful approach to minimizing project risk for the emerging GIF concepts.

Enabling Safeguards by Design: Robust safeguards are essential to the PR&PP characteristics of all of the emerging GIF designs. In conjunction with the PRPPWG effort with the SSCs, the PRPPWG will maintain cognizance of technology developments and good practices that would foster safeguards-by-design in the GIF designs.

Small Modular Reactors: To the extent that it is relevant to GIF designs, the PRPPWG will maintain cognizance of this area and enable the incorporation of robust PR&PP features in the SMRs. The emergence of SMRs as a major design consideration in the second decade of GIF, with potential impact on the GIF designs themselves (particularly in scaling of designs, as required) indicates the importance of cross-cutting evaluation methodologies that are as generic as possible. The flexibility that allows non-GIF users to apply the PR&PP methodology also maintains the methodology's relevance to GIF design teams as specifications are modified.

IAEA/INPRO: The PRPPWG will continue to coordinate with IAEA in areas of mutual interest (an immediate area of coordination being the PROSA project of INPRO). In general, the PRPPWG will maintain cognizance of developments in safeguards concepts and approaches, and assess and respond to any potential impact on the relevance of the PR&PP methodology.

Continued interaction between the PRPPWG and the other GIF crosscutting groups: Coordination with the RSWG and with the Economics Modeling Group should be pursued to assure effective implementation of approaches in the GIF design. As noted earlier, the aggregation of PRPPWG and RSWG represents an implementation of the "3S" approach of the IAEA.

VII. CONCLUSION

The PRPPWG has developed a mature evaluation methodology that is not only ready to assist GIF SSC's in making informed design choices based on PR&PP principles, but also represents the most comprehensive publicly available PR&PP evaluation methodology and can similarly inform the design process of any new nuclear technology.

The PR&PP methodology is aligned with international efforts to improve the effectiveness of and efficiency of safeguards. It represents an enabling tool for Safeguards by Design, and, in conjunction with the Reactor Safety Working Group of GIF, a natural manifestation of the so-called "3S" integration of Safety, Security, and Safeguards within the culture of nuclear technology design.

The PRPPWG will continue to work with the SSCs to implement pilot applications of the PR&PP methodology, as well as maintaining cognizance of international developments and engagement of other groups within the international non-proliferation community. The PR&PP methodology will be maintained as necessary to retain its relevance and applicability to the development of new and emerging nuclear systems, primarily within GIF but also to the broader nuclear community.

ACKNOWLEDGEMENTS

The efforts and ideas of the many members of the PR&PP working group over the past decade is the foundation of this summary paper. For a list of contributors to Revision 6 of the Methodology Report, see http://www.gen-4.org/PDFs/GIF_PRPPEM_Rev6_FINAL.pdf, p.3. We are grateful to J. Cazalet, G. Cojazzi, and J. Sprinkle for providing valuable comments and suggestions on this manuscript.

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COST ESTIMATION WITH G4-ECONS FOR GENERATION IV REACTOR DESIGNS

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I. INTRODUCTION

The Economic Modeling Working Group (EMWG) was created by Generation IV International Forum (GIF) early in the Generation IV process. 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.

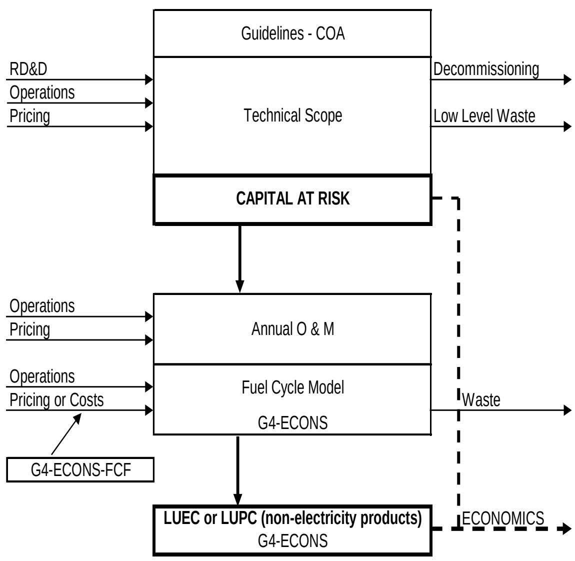
The GIF Cost Estimating Methodology has been developed and tested by the EMWG. It consists of 1) The Generation IV Cost Estimating Guidelines and 2) a software package, G4-ECONS, to facilitate the implementation of the Guidelines.

In this paper, the results of an application of G4-ECONS on the Generation IV systems is presented. The study was part of an independent determination and assessment of plant design characteristics of future nuclear reactor designs and their associated fuel cycles. All six Generation IV designs have been assessed and compared to a reference Generation III design.

II. GIF COST ESTIMATING GUIDELINES AND G4-ECONS

The GIF Cost Estimating Guidelines provide a comprehensive approach for assessing the performance of Generation IV systems in achieving the established economic goals [1, 2]. The methodology may be used to assess the cost structure of the Generation IV systems in comparison with Generation III systems, identify cost drivers and possibilities for design improvement in this regard. The Guidelines provide detailed processes for developing the total capital investment 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 [2]



The central feature of the methodology is the comprehensive Code of Accounts. The Code of Accounts (COA) provides a disciplined structure for capturing and categorizing all appropriate costs in the development of a consistent system cost estimate.

Because the Generation IV systems will for some time be in varied states of development and maturity, a “top down” approach with scaling factors is considered for the cost estimation.

To facilitate implementation of the Cost Estimating Guidelines, the EMWG developed an EXCEL based spreadsheet package, G4-ECONS. For this study, G4-ECONS version 2.0 was used [3, 4]. Levelized unit electric cost is calculated, as well as its components capital, O&M and fuel cycle costs.

III. SELECTED GENERATION IV REACTOR DESIGNS AND FUEL CYCLES

Reactor design was not part of the study, therefore existing design data have been used from other studies. For each Generation IV system [5], a design basis has been selected. Also, as most data were available for near term application of the system, an HTR design has been selected instead of a "real" VHTR (with outlet temperatures above 1 000°C), and the fast reactor fuels have been selected as MOX based, instead of the more advanced nitride, carbide or metal fuels.

The following reactor designs have been selected to represent the six Generation IV concepts in this study:

- HTR: HTR-PM. This is a twin pebble bed high temperature reactor, based on the German HTR Modul design [6]. It is currently under construction in China. The fuel is enriched uranium oxide in the form of coated particles in pebble fuel.
- SFR: European SFR. This is a sodium cooled fast reactor based on design of European Commission R&D project ESFR [7], Phénix, Superphénix, and EFR design experience. The fuel is fast reactor MOX fuel.
- LFR: European LFR. This is a lead cooled fast reactor based on the European Commission R&D projects ELSY [8] and its follow-up project LEADER. The fuel is fast reactor MOX fuel.
- GFR: European GFR. This is a gas cooled fast reactor based on the European Commission R&D projects GCFR [9] and its follow-up project GoFastR. The fuel is fast reactor MOX fuel.
- SCWR: European SCWR. This is a super-critical water reactor based on the European Commission R&D projects HPLWR1 and its follow-up project HPLWR2 [10], and the Japanese design JSCWR. It was largely based on the German Gundremmingen BWR. The fuel is LWR MOX fuel.
- MSR: US American MSBR. This is a thermal molten salt breeder reactor, designed by Oak Ridge National Laboratory around 1970 [11]. The fuel is thorium molten salt.

These reactor designs are being compared with a reference Generation III design, in order to allow intercomparison of the Generation IV designs. The reference Generation III design selected is the EPR as under construction in Finland, France and China, by the French vendor Areva.

The selected designs have very much different power levels, as illustrated by Figure 2. In addition, the thermal conversion efficiency figures used in this study have been collected from the available literature, and shown in Figure 3.

Figure 2: Power levels of the six selected Generation IV designs and the reference Generation III design

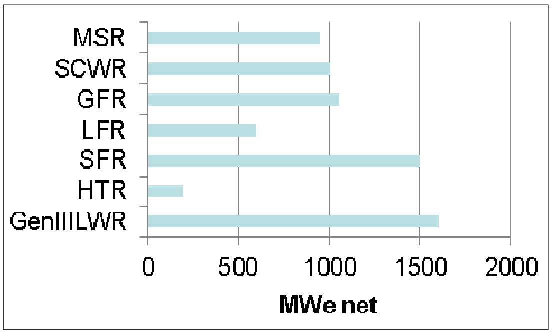
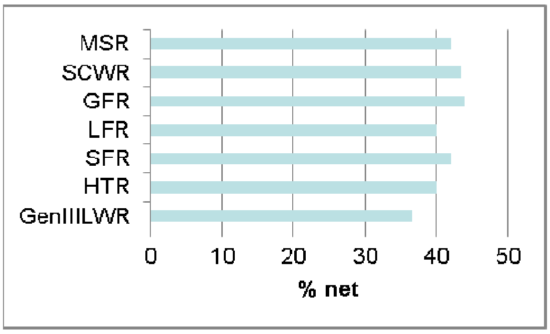


Figure 3: Efficiency figures of the six selected Generation IV designs and the reference Generation III design



IV. COST ESTIMATE RESULTS

Construction costs have been estimated by scaling from known cost distributions and adaptation by expert judgement. This methodology is elaborately described in [12].

First, the relative distributions of the costs for different reactor types are collected from literature sources for the construction cost accounts 2 to 4. Then, following the GIF EMWG COA [1], the overnight construction cost components of the different reactor types are estimated relative to the construction costs of a reference plant which are put to 100%.

One of the main considerations for each code of account is to scale reactor data to the reference plant data with the net electric (or thermal) power. The scaling relationship in the equation below has been employed, using different scaling factors for reactors employing a small (up to 20%) and a large difference in net electric (or thermal) power [13].

$$Cost_{new} = Cost_{ref} \left(\frac{Power_{new}}{Power_{ref}} \right)^{a}$$

in which "Cost$_{new}$" and "Cost$_{ref}$" are the costs of the considered plant and the reference plant respectively, "Power$_{new}$" and "Power$_{ref}$" are the power levels of the considered plant and the reference plant respectively, and "a" is the scaling factor.

Besides scaling to power level, other considerations may lead to increase or decrease certain accounts with respect to the accounts of the reference design. For example, the reactor vessel and other reactor plant equipment may be expected to pay a larger contribution to the overnight construction costs for Generation IV reactors than for Generation III reactors because of the application of more expensive materials which can withstand elevated temperatures and more demanding coolants. Also the vessel size, related to power density, and pressure may be different. Other considerations include:

- Space requirements.
- Containment size.
- Application of passive safety systems.
- Need for an intermediate circuit.
- Complex fuel handling in HTR, the FR and MSR.
- Use of chemically highly reactive sodium as coolant in SFR.
- Use of Rankine vs. Brayton cycle.

In this way, construction costs and their distribution have been determined for the six systems in relation with the reference Generation III design as indicated in Figures 4 and 5. The ranges indicated in Figure 4 are derived from the contingency figures from INL [13], increased by a factor 1.5 because of these figures come from an older (1995) study on the ALMR, a relatively proven SFR design compared to most other Generation IV concepts.

Figure 4: Overnight construction cost ranges

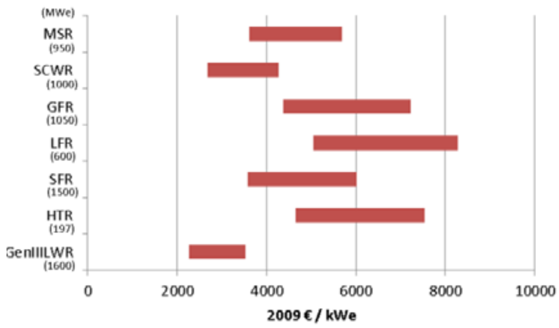
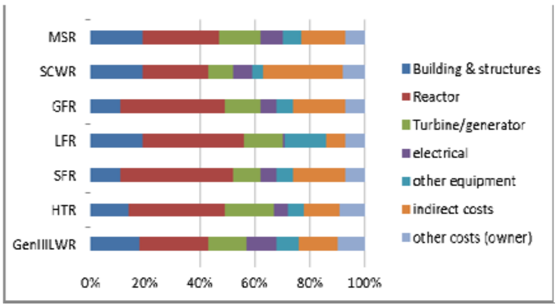


Figure 5: Distribution of the costs over the main cost components



The relatively high costs of the LFR can be attributed to the small power level compared to the SFR while in the current assessment the benefits of small reactor designs allowing for modular construction and increasing the learning curves are not considered. For the MSR, it should be known that the fuel processing plant equipment costs are included in the construction costs.

Also cost ranges have been determined for the Operation and Maintenance costs, as shown in Figure 6. They are based on various literature sources.

The Fuel cycle costs are shown in Figures 7 and 8, divided into front-end and back-end costs. When estimating costs for Generation IV reactor fuel cycles, accounting for non-conventional fuels is needed. Compared to the reference Generation III LWR, the pebble bed HTR has a large number of fuel assemblies, a small amount of heavy metal mass per fuel assembly, a small amount of elements per reload and a short time between refuellings (simulating continuous on-line reloading). A similar definition has been used for the MSR, with continuously reloaded liquid fuel simulated by "very small fuel assemblies". A workaround simulation thorium fuel had to be introduced, as the thorium fuel cycle was not included yet in this version of G4-ECONS. In the model, thorium has been taken for uranium, and uranium for plutonium.

Figure 6: Operation and maintenance cost ranges

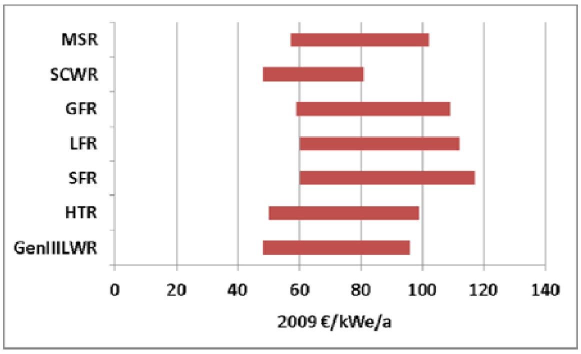
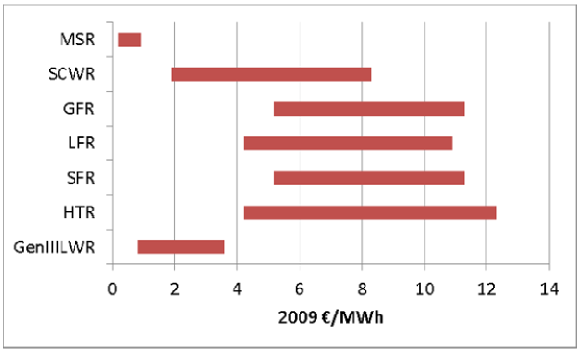
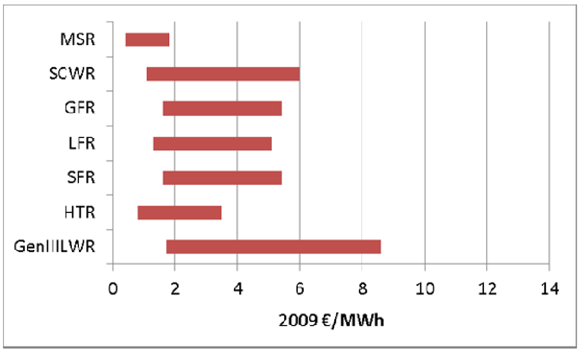


Figure 7: Front-end fuel cycle cost ranges



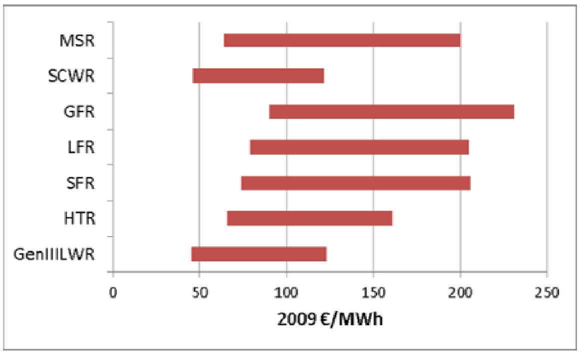
It can be observed that all Generation IV designs except the MSR show increased front-end costs due to more costly fuel fabrication. The MSR costs are low because of the absence of fuel element fabrication and dismantling. Also, the MSR fuel processing costs are in the investment costs, not in the fuel cycle costs. On the other hand, the back-end costs for all Generation IV designs are low compared to the Generation III reference design.

Figure 8: Back-end fuel cycle cost ranges



Taking construction costs, O&M and fuel cycle costs together, the levelized electricity generating cost (LUEC) ranges have been calculated, as shown in Figure 9. The discount rate has been set on 10%.

Figure 9: Levelized electricity generating cost ranges



It can be observed from Figure 9 that only the SCWR has electricity generating costs comparable to the reference Generation III LWR; all other designs have higher generating costs. On the other hand, the ranges are wide, and are overlapping for all systems. Also, special cost-influencing features for specific reactor systems, like cogeneration for the HTR, or invulnerability against very high uranium prices for the fast reactor systems, have not been investigated.

V. CONCLUSIONS

The Generation IV Cost Estimating Methodology appeared to be very suitable to produce a consistent cost evaluation of the six Generation IV Systems. All selected Generation IV reactor systems have higher specific construction costs, compared to the reference Generation III LWR design. Also the front-end fuel cycle costs are mostly higher, a price being paid to decrease the back-end fuel costs. When regarding electricity as the only product, and within the current circumstances like uranium price and discount rate, only the SCWR equals the electricity generating cost range of the reference Generation III LWR, all other systems have higher generating costs. However in general, the cost results still come with large uncertainties, and should be regarded as preliminary. The economic assessments can then be improved and uncertainties reduced as the designs mature.

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SAFETY DESIGN CRITERIA FOR GENERATION IV SODIUM-COOLED FAST REACTOR SYSTEM

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ABSTRACT

In the framework of the GIF, an effort on development of the "Safety Design Criteria [SDC]" for the Sodium-cooled Fast Reactor [SFR] system was initiated in 2011 with the intent of completion in two years. The objectives of the SDC are to provide the reference criteria of the safety designs of structures, systems and components (SSC) of the SFR system, where the criteria are clarified systematically and comprehensively consistent with the GIF's basic safety approach and with the aim of achieving the safety and reliability goals defined in the GIF Roadmap. The SDC intends to maintain the basic structures of texts in the IAEA SSR 2/1. The contents of the SDC are grouped mainly into four parts: In the first part, the formulation principles of the SDC and the key viewpoints to interpret the GIF's safety goals and basic safety approach to the criteria for the safety design are described. In the second part, 83 criteria for the overall plant design and specific SSC designs are described in sequence. The third part is the Glossary which includes specific terminologies of Generation IV reactor systems and SFR system. The fourth part is the appendix which includes key items to understand examples of the SFR system configuration and technical backgrounds related to the SFR safety characteristics. The SDC would be disseminated to not only the GIF community but also to international technical entities, and expected to provide guidance for SFR designs at international level.

I. INTRODUCTION

Development of the "Safety Design Criteria [SDC]" for the GIF Sodium-cooled Fast Reactor [SFR] system was proposed in October 2010 at the GIF Policy Group meeting, and the Terms-of-Reference for establishment of the SDC task force [TF] was approved in May 2011. The SDC TF was started in July 2011, with the target of summarizing/consolidating the SDC within ca. 2-year schedule. The SDC TF members consist of the representatives from many organizations, which are CIAE (China), CEA (France), JAEA (Japan), KAERI, KINS (Republic of Korea), IPPE (Russia), ANL, INL, ORNL (United States of America), EC and IAEA. Four TF meetings were held to develop and revise the SDC. A number of inputs, comments and counterproposals from the SDC TF members and other GIF entities (i.e. the Risk & Safety Working Group [RSWG], SFR System Steering Committee, Senior Industry Advisory Panel, and Expert/Policy Groups [EG/PG]) were discussed and incorporated in the SDC.

The objectives of the SDC is to present the reference criteria of the safety designs of structures, systems and components [SSC] of the SFR system, where the criteria are clarified systematically and comprehensively, for adopting the GIF's basic safety approach established by the RSWG, with the aim of achieving the safety and reliability goals defined in the GIF Roadmap by the PG.

The contents of the SDC are grouped mainly into four parts: The first part is the Chapters 1 and 2 where the formulation principles of the SDC and the key viewpoints to interpret the GIF's safety & reliability goals and basic safety approach to the criteria for the safety design are described. The second part is from Chapters 3 to 6, where eighty-three criteria of the overall plant design and specific SSC design are described in sequence. The format of the second part is consistent with that of the IAEA SSR 2/1 [1] for the convenience of not only SFR concept developers under the GIF but also other R&D

and regulatory entities interested in the SFR technology. The SDC maintains the basic text in the SSR 2/1 as it applies to Gen IV SFR systems. The third part is the glossary and the fourth part is the appendix which includes examples of key items of the SFR system configuration and technical background to understand better the SFR safety characteristics.

II. FUNDAMENTAL APPROACH TO THE SDC ESTABLISHMENT

II.A. SDC position in safety standard hierarchy

As the Gen IV SFR development is progressing toward conceptual design stage at least in some member states, harmonization of safety principles is increasingly important for realization of enhanced safety features common to SFR systems. From the viewpoint of the safety standard hierarchy shown in Figure 1, the upper level safety standards for the Gen IV system have so far been outlined in:

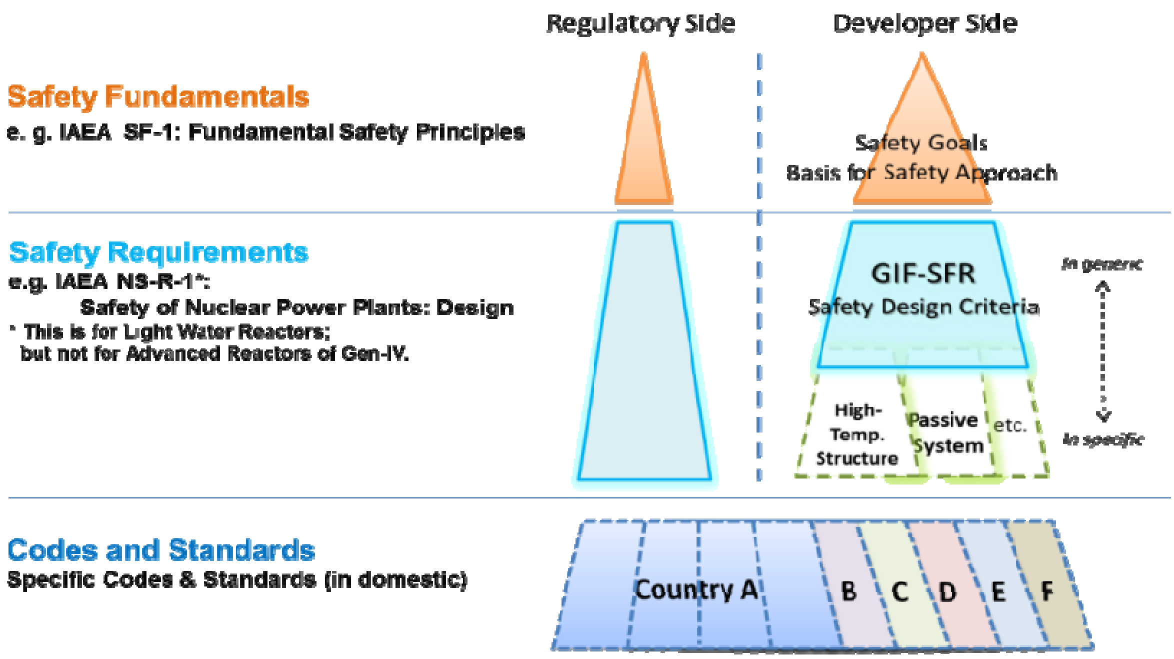
- Safety and reliability goals for "Generation IV Nuclear Energy Systems under the GIF Roadmap" [2] [GRM] by the PG.
- "Basis for safety approach for design & assessment of Generation IV Nuclear Systems" [3] [BSA] by the RSWG.

The design requirements of the SFR systems have also been clarified in:

- "SFR System Research Plan" [4] [SRP] by the SFR System Steering Committee.

However, it can be recognized from Figure 1 that there is a large gap between the "upper level" safety standards and the "base level" country-specific codes & standards which will be referenced and used for designing Gen IV nuclear systems. The GIF SFR SDC is intended to cover that gap and provide the set of general criteria for the design of SSC of the Gen IV SFR system that the safety level meets the Gen IV safety goals and the safety approach follows those established in the BSA.

Figure 1: Hierarchy of safety standards, and relation between regulatory and development sides



II.B. GIF's safety goals and safety approach, and SFR design tracks

The GIF Roadmap set up three goals summarized below:

- SR-1: Excel in Operational Safety and Reliability.

This goal notes on safety and reliability during normal operation and likely operational events that assume forced outages.

- SR-2: Very low likelihood & degree of reactor core damage.

This goal notes on minimizing the frequency of initiating events that could lead to core damage and inclusion of design features for controlling and mitigating any initiating events without causing core damage.

- SR-3: Eliminate the need for offsite emergency response.

This goal notes on safety architecture to manage and mitigate severe plant conditions, for minimizing the possibility and the amount of releases of radiation.

The BSA report proposes a technology-neutral basic safety approach for application of the Defence-in-Depth [DiD] philosophy and risk-informed safety approach to design of the nuclear system so that safety function is built in (and not be added-on). The BSA aims at achieving high level safety meeting to GIF's safety and reliability goals.

The SFR System Research Plan (SRP) describes three system configurations as shown in Table 1:

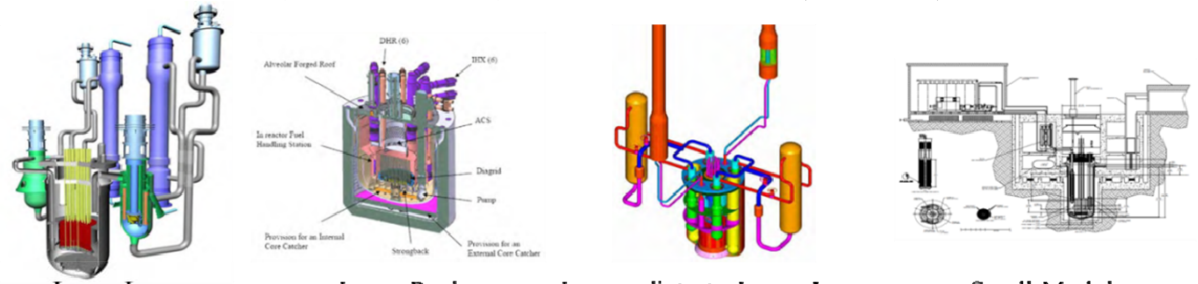
- A large size (600 to 1 500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors.
- An intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal fuel.
- A small size (50 to 150 MWe) modular-type reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel.

The four SFR design tracks of JSFR, KALIMER, ESFR, and SMFR as shown in Figure 2, as examples. The other system configuration options are arranged on coolant system (primary and secondary coolant system utilizing sodium coolant), balance-of-plant (water/steam cycle, and alternative concept of supercritical CO_2 cycle), and fuel (MOX, metal, and others).

Table 1: System configuration options of the SFR system under GIF

System structure	Loop-type, pool-type, small modular
Electric output	50-2 000 MWe
Coolant system	Primary and secondary (intermediate) coolant system utilizing sodium coolant
BOP system	Water/steam cycle (alternative concept: supercritical CO_2 cycle)
Fuel	MOX, metal, others

Figure 2: Four design tracks of the GIF SFR (as of 2011)



Large Loop | Large Pool | Intermediate-to-Large Loop | Small Modular

II.C. Basic scheme to outline the SDC

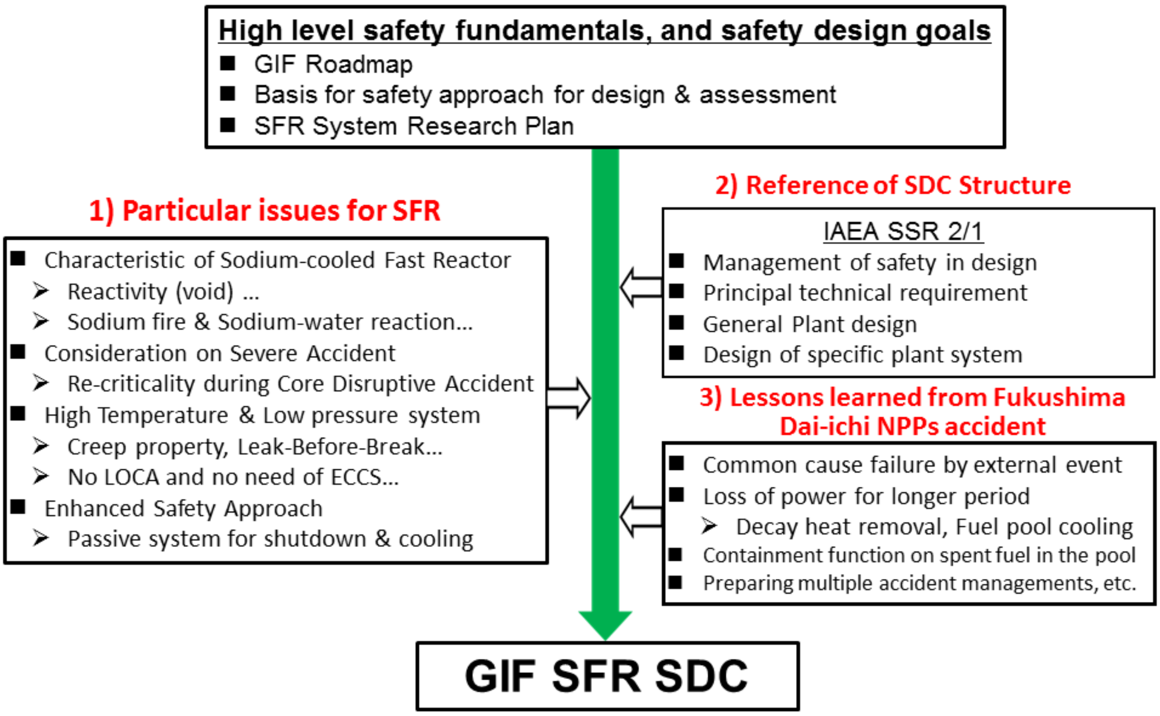
At the beginning of the SDC establishment, three basic themes were identified as shown in Figure 3. The first is that the safety level for Gen IV reactors should be achieved, the second is that the specific features of SFR system should be considered, and the third is that the latest knowledge should be incorporated as they become available – for example, safety-related R&D results on innovative technologies and lessons learned from the TEPCO's Fukushima Daiichi [F1] accident.

Three policies were established as well for the SDC formulation:

- Policy on goals

These criteria are intended to apply Gen IV SFR systems and they reflect the goals documented in the GIF roadmap. Due to large participation, however, it can also be viewed as the latest international opinion on the safety criteria that should be taken into account for the SFR design and licensing.

Figure 3: Basic scheme to outline the SDC



- Policy on descriptions

Attention is given to the GIF safety goals/approaches, and the criteria providing performance targets are described in greater depth. The basis of SFR-specific criteria, including the reason and background, are provided for further clarification.

- Policy in definitions and terminology

The definitions of the DiD and plant states are based on IAEA INSAG-12 [5] and SSR-2/1 as shown in Figure 4. The IAEA SSR 2/1 is also considered as a reference document regarding the basic approach on safety and comprehensive formulation of the contents. The safety-related terms for the SDC are basically the same as the ones defined in the IAEA Safety Glossary [6], and additional definitions in the SDC are made as needed for specific terms related to the Gen IV SFR systems.

III. CONTENTS OF SAFETY DESIGN CRITERIA

III.A. SDC structure and criteria

The table of contents of the SDC is shown in Table 2. The first part, Chapters 1 and 2, presents the formulation principles of the SDC and the key viewpoints on how to relate the GIF's safety goals and basic safety approach to the criteria for the safety design. The second part, Chapters 3 to 6, presents eighty-three criteria and 206 paragraphs of the overall plant design and specific SSC designs in sequence.

Figure 4: Defence-in-depth and plant state definitions in the SDC (Based on the IAEA INSAG-12 and SSR 2/1)

DiD Levels				
Level 1	Level 2	Level 3	Level 4	Level 5
plant states (considered in design)				Off-site emergency response (out of the design)
Normal operation	AOO	DBA	DEC	
Operational states		Accident conditions		
Normal operation	Anticipated operational occurrences	Design basis accidents	Design extension conditions (including Severe Accident conditions)	

Table 2: Table of contents of the SDC

1. INTRODUCTION 1.1 Background and Objectives 1.2 Principles of the SDC formulation 2. SAFETY APPROACH TO THE SFR AS A GENERATION-IV REACTOR SYSTEM 2.1 GIF Safety Goals and Basic Safety Approach 2.2 Fundamental Orientations on Safety 2.2.1 DiD and Plant States 2.2.2 Relationship among plant states, probabilistic and deterministic approaches 2.2.3 Utilization of passive safety features 2.3 Safety approach of the Generation-IV SFR systems 2.3.1 Target SFR Systems 2.3.2 Approach based on basic characteristics of the SFR 2.3.3 Approach to safety during normal operation, AOO, DBA, and DEC specific to the SFR 2.3.4 Lessons Learned from TEPCO's Fukushima Dai-ichi NPPs Accidents 3. MANAGEMENT OF SAFETY IN DESIGN 4. PRINCIPAL TECHNICAL CRITERIA 5. GENERAL PLANT DESIGN 5.1 Design Basis 5.2 Design for Safe Operation over the Lifetime of the Plant 5.3 Human Factors 5.4 Other Design Considerations 5.5 Safety Analysis	6. DESIGN OF SPECIFIC PLANT SYSTEMS 6.1 Overall Plant System 6.2 Reactor Core and Associated Features 6.3 Reactor Coolant Systems 6.4 Containment Structure and Containment System 6.5 Instrumentation and Control Systems 6.6 Emergency Power Supply 6.7 Supporting Systems and Auxiliary Systems 6.8 Other Power Conversion Systems 6.9 Treatment of Radioactive Effluents and Radioactive Waste 6.10 Fuel Handling and Storage Systems 6.11 Radiation Protection GLOSSARY APPENDIX (A) Definitions of Boundaries of SFR systems (B) Guide to Design Extension Conditions (C) Guide to Practical Elimination of accident situations (D) Guide to Utilization of Passive Measures (E) Approach to Extreme External Events

Criteria 1-3 are related to "Management of Safety in Design".

Criteria 4-12 are related to "Principal Technical Criteria". Criterion 7: "Application of defence in depth" has been updated from the SSR 2/1 for including the safety approach as for the Gen IV nuclear system.

Criteria 13-42 are related to "General Plant Design" where "Design Basis" (including Internal and External hazards), "Design for Safe Operation over the Lifetime of the Plant", "Human Factors", "Other Design Considerations", and "Safety Analysis" (including Deterministic & Probabilistic approaches) are described. In this part, Criterion 19: "Design Basis Accident [DBA]", Criterion 20: "Design Extension Condition [DEC]" and Criterion 31: "Ageing management" have been updated for including the safety approach as for the Gen IV nuclear system. Criterion 17: "Internal and External hazards" was extended for the SFR system to include the lessons learned from the F1 accident.

Criteria 42bis-82 are related to "Design of Specific Plant Systems" where "Overall Plant System", "Reactor Core and Associated Features", "Reactor Coolant Systems", "Containment Structure and Containment System", "Instrumentation and Control Systems", "Emergency Power Supply", "Supporting Systems and Auxiliary Systems", "Other Power Conversion Systems", "Treatment of Radioactive Effluents and Radioactive Waste", "Fuel Handling and Storage Systems" and "Radiation Protection" are included. This chapter deals with SFR specific SSCs and many additions/updates have been incorporated for considering SFR characteristics such as "low pressure and high temperature coolant conditions", "reactivity characteristics of fast spectrum system", "reactivity feedbacks", and "sodium coolant and chemical reactivity." The lessons learned from the F1 accident is also included, for example, in terms of the requirements on instrumentations under DEC condition and long-term cooling of spent fuel pool.

In comparison to the SSR-2/1, the SDC includes twenty modified criteria, two newly added criteria, and one deleted criterion. Sixty criteria of the SSR-2/1 are retained essentially unchanged. Texts of forty-eight paragraphs are modified, eighteen are added, two are deleted, and 138 are unchanged.

The third part, Glossary, includes specific terminologies of Gen IV reactor systems and SFR system. The fourth part, Appendix, includes key items to understand examples of the SFR system configuration and technical backgrounds related to the SFR safety characteristics, e.g. coolant/containment boundaries of the SFR systems.

III.B. Examples of the SDC

III.B.1. Related to Gen IV safety goal

Criterion 20: "Design Extension Conditions" is an example related to Gen IV safety goal to describe difference on the safety level to the Generation III nuclear system. In paragraph 5.31 of the IAEA SSR 2/1, it is noted:

> The design shall be such that design extension conditions that could lead to significant radioactive releases are practically eliminated. If not, for design extension conditions that cannot be practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures.

In the GIF SFR SDC, it is noted as below:

> The design shall be such that design extension conditions that could lead to significant radioactive releases are practically eliminated by means of measures for prevention and/or mitigation of severe core degradation and serious fuel failures during fuel handling and storage.

Practical elimination of significant radioactive releases is required in both SSR 2/1 and SDC, but the measures for prevention/mitigation of severe accidents are explicitly specified (as noted with underline) in the SDC.

III.B.2. Related to SFR specific SSCs

Criterion 46: "Reactor shutdown" is an example related to SFR specific criterion in relation to prevention of severe accidents. In paragraph 6.9 of the IAEA SSR 2/1, it is noted:

> The means for shutting down the reactor shall consist of at least two diverse and independent systems.

In the GIF SFR SDC, it is noted:

> The means for shutting down the reactor shall consist of at least two diverse and independent systems. For design extension conditions, passive or inherent reactor shutdown capabilities shall be provided to prevent severe core degradation and to avoid re-criticality in the long run.

At least two means for reactor is required in both. The point of difference is that additional shutdown capability using passive or inherent feature is included for the DEC for prevention of severe accidents and to avoid re-criticality for the long term.

III.B.3. Related to lesson learned from F1 accident

Criterion 80: "Fuel handling and storage systems" is an example related to lesson learned from F1 accident. In paragraph 6.67 of the IAEA SSR 2/1, it is noted:

> The fuel handling and storage systems for irradiated fuel shall be designed: (a) to permit adequate removal of heat from the fuel in operational states and in accident conditions;

> In the GIF SFR SDC, it is noted as below:

> The fuel handling and storage systems for irradiated fuel shall be designed: (a) to permit adequate removal of heat from the fuel and monitoring its status in operational states and in accident conditions including the long-term loss of all AC power supplies.

The point is that "status monitoring" under accident conditions (including DEC) and heat removal under "long-term loss of all AC power supplies" are explicitly included.

IV. KEY SAFETY APPROACH IN THE SDC

IV.A. Safety approach in relation to plant states

Gen IV reactor systems aim at achieving a higher safety level than that of Generation III systems. As for the safety approaches for normal operation, anticipated operational occurrence and DBA, it is pointed out that feedback on "operation/accident experience" and "maintenance/repair experience" is important. High reliability systems will be achieved by improvements and developments obtained from operational experience of current reactors, by the enhancement of safety margins through the introduction of new technologies, and by the improvement of inspection technology capable of detecting conditions that could lead to failures. As for the safety approach for DEC, it is pointed out that providing practical measures for managing DEC is important in order to prevent their occurrences and/or mitigate their consequences. Due consideration on the common cause failures shall be taken into account in the safety design. Applying passive design measures, by utilizing/enhancing favourable safety features specific to the SFR system, will be encouraged.

The identification/selection of DBA and DEC will be based on the combined use of: 1) "Deterministic approach" based on fundamental characteristics of the reactor system supplemented by probabilistic analysis as needed, 2) "Operation experience" & "External event experience", and 3) "Licensing experience".

IV.B. Utilization of passive safety features

Provisions of well-balanced design measures are necessary by using an appropriate combination of active and passive safety systems in order to enhance safety against a number of wide-ranging events. For DBA, it is important to well characterize the safety SSCs and to enhance the reliability of the safety systems. For DEC, however, it is possible to ensure diversity with different operation principles, without further increasing the redundancy of the measures already applied for DBA. Using passive safety and inherent safety features of the design will allow termination of accidents or mitigation of consequences of a DEC, even in postulated failure of active safety systems.

IV.C. Safety features of sodium-cooled fast reactor

The SDC defines the safety approach based on basic characteristics of the SFR as follows.

IV.C.1. Core and fuel characteristics

One characteristic of an SFR is that the core fuel is not in the most reactive configuration, and that it is possible to have a positive void reactivity in the centre area of the reactor core. Due considerations on these characteristics are necessary for the reactor core design to have an inherent reactivity feedback to control reactor power in the operational states and to prevent the re-criticality leading to significant mechanical energy release during a hypothetical core disruptive accident. The criteria included in the SDC reflect these requirements.

IV.C.2. Physical and chemical properties of sodium coolant

A positive feature of sodium is that it has a high thermal conductivity. The boiling temperature is 883°C at atmospheric pressure, significantly higher than the typical average core outlet temperature of 500-550°C. Hence, decay heat removal is possible using natural circulation due to the favourable coolant characteristics.

However, sodium is chemically active and it is therefore necessary to manage sodium leaks (sodium fire on contact with air and reaction with water or concrete). It is also necessary to introduce a secondary coolant system so that a sodium leak does not affect the safety of the reactor. Sodium is opaque and freezes at room temperature (having a melting point of 98°C). Hence, due consideration of this high melting point of sodium is necessary in the design of the SSCs when addressing capabilities for inspection, maintenance and repairing. The SDC includes statements to reflect the requirements related to these coolant characteristics.

IV.C.3 Material usage environment

As an SFR operates at a relatively high temperature compared to an LWR (e.g. the coolant temperature range is around 300-600°C) and in high fast neutron fluence conditions, due consideration of creep and radiation effects on fuel and structural materials is necessary. Hence, due consideration on thermal striping and thermal shock is necessary as specified in the SDC.

IV.C.4. Operation under low pressure condition

As an SFR is operated under low pressure conditions, coolant leakage does not lead to the type of loss of coolant accident anticipated in an LWR with depressurization, coolant boiling and the loss of cooling capability. Therefore, an emergency core cooling systems for coolant injection under high and low pressure conditions, as used in the LWR, are not necessary in an SFR. The only requirements for SFR core cooling are the maintenance of the sodium coolant level above the reactor core in the reactor vessel along with sufficient heat removal capability. These requirements are also reflected in the SDC.

V. CONCLUDING REMARKS

The draft SDC is now released for internal reviews under the auspices of GIF. A number of inputs, comments and counterproposals in the SDC TF and from other GIF entities have already been incorporated. In its current form, the SDC systematically and comprehensively clarifies the criteria for the SSCs' safety designs, and it is expected that the SDC would be disseminated and utilized for SFR design at international level in interactions with IAEA and MDEP.

Safety improvement comes from continuous efforts for updating safety designs based on the new safety technology and recent knowledge related to the operation experiences and R&D outcomes. In this sense, the SDC will be continuously updated as necessary, by including constructive feedbacks from the GIF community and all the international technical entities.

During the course of SDC development, the TF discussions contributed significantly to the establishment of common understanding of safety issues for Generation IV SFR. As a future work, a similar effort towards the development of safety design guideline (SDG), which is more detailed recommendation and guidance to safety design, will be considered through a similar task force arrangement.

ACKNOWLEDGEMENTS

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NOMENCLATURE

BSA	Basis for safety approach
DBA	Design Basis Accident
DEC	Design Extension Condition
DiD	Defence-in-Depth
EG/PG	GIF Expert Group & Policy Group
ESFR	European Sodium-Cooled Fast Reactor
F1	Fukushima Dai-ichi
GIF	Generation IV International Forum
GRM	GIF Roadmap
IAEA	International Atomic Energy Agency
JSFR	Japan Sodium-Cooled Fast Reactor
KALIMER	Korea Advanced Liquid Metal Reactor
LWR	Light Water Reactor
RSWG	GIF Risk & Safety Working Group
SDC	Safety Design Criteria
SFR	Sodium-Cooled Fast Reactor
SMFR	Small Modular Fast Reactor
SSC	Structures, systems and components
TEPCO	Tokyo Electric Power Company, Inc.
TF	Task Force

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CURRENT STATUS OF COLLABORATION FOR GIF SODIUM-COOLED FAST REACTOR SYSTEM

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ABSTRACT

Collaborations for the GIF Sodium-Cooled Fast Reactor (SFR) system encompass research and development in the areas of advanced fuels, safety approach, in-service inspection, Phénix, Monju and possibly CEFR and BN-800 tests, components, advanced energy conversion systems, and materials, codes and standards. These collaborative activities are being conducted under the Project Arrangements of Advanced Fuel (AF), Global Actinide Cycle International Demonstration (GACID), Component Design and Balance-of-Plant (CDBOP), and Safety and Operation (SO) projects. According to these efforts, most objectives have been well followed up within the GIF framework except for "materials, codes and standards", and most challenging technology gaps have been identified by projects as well. The System Integration and Assessment Project Arrangement has been prepared and will start, after completion of the signing process, in order to integrate and assess the results of R&D work conducted under the R&D Project Arrangements. The CDBOP and SO Project Plans have been updated and are in the signature process.

I. INTRODUCTION

Sodium-Cooled Fast Reactor (SFR) system features a fast-spectrum reactor and closed fuel recycle system. The primary mission for the SFR is improved resource utilization, management of high-level wastes and, in particular, management of plutonium and other actinides. With innovations to reduce capital cost, the mission can extend to electricity production; given the proven capability of sodium reactors to utilize almost all of the energy content in the natural uranium versus a few percent utilized in thermal spectrum systems.

The Sodium-Cooled Fast Reactor uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system.

Plant size options under consideration range from small modular reactors (50 to 300 MWe) to larger plants (up to 2 000 MWe). The outlet temperature is 500-550°C for the options, which affords the use of the materials developed and proven in prior fast reactor programs.

The SFR closed fuel cycle enables regeneration of fissile fuel and facilitates management of minor actinides. However, this requires that recycle fuels be developed and qualified for use. Important safety features of a Generation IV system include a long thermal response time, a reasonable 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 supercritical carbon dioxide are considered as working fluids for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR is aimed to be economically competitive in future electricity markets. In addition, the fast neutron spectrum greatly extends the uranium resources compared to thermal reactors. The SFR is considered to be the nearest-term deployable Generation IV system.

Much of the basic technology for the SFR has been established in former fast reactor programs, and further confirmed by the Phénix end-of-life tests in France, the restart of Monju in Japan, the lifetime extension of BN-600, and the startup of the China Experimental Fast Reactor.

Although the SFR system is dedicated to actinide management, if enhanced economics for the system can be realized, it can also be used for the production of electricity and heat. The SFR is an attractive energy source for nations that desire to make the best use of limited nuclear fuel resources and manage nuclear waste by closing the fuel cycle.

Fast reactors hold a unique role in the actinide management mission because they operate with high energy neutrons that are more effective at fissioning transuranic actinides. The main characteristics of the SFR for actinide management mission are:

- Consumption of transuranics in a closed fuel cycle, thus reducing the radiotoxicity and heat load to facilitate waste disposal and geologic isolation.
- Enhanced utilization of uranium resources through efficient management of fissile materials and multi-recycle.

High level of safety achieved through inherent and passive means also allows accommodation of transients and bounding events with significant safety margins.

The reactor unit can be arranged in a pool layout or a compact loop layout. Three options are considered in the GIF SFR System Research Plan:

- A large size (600 to 2 000 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors as shown in Figure 1.
- An intermediate-to-large size (300 to 2 000 MWe) pool-type reactor with oxide or metal fuel as shown in Figures 2 and 3.
- A small size (50 to 150 MWe) modular-type 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 as shown in Figure 4.

Figure 1: JSFR (loop-configuration SFR)

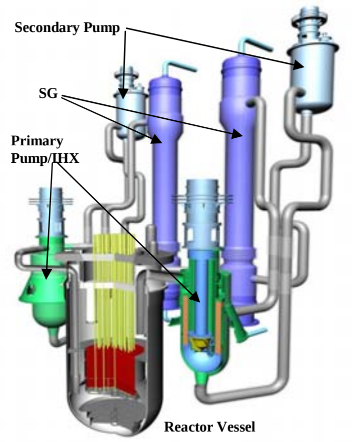


Figure 2: ESFR (pool-configuration SFR)

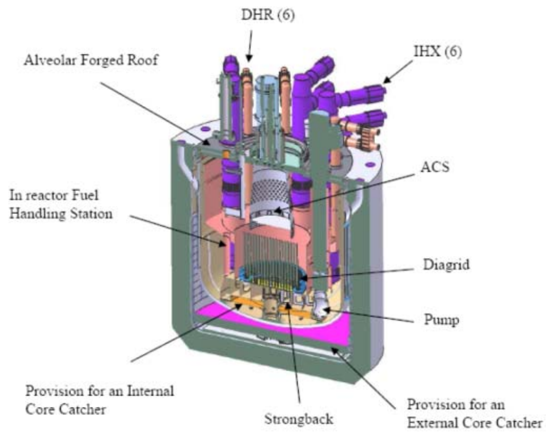
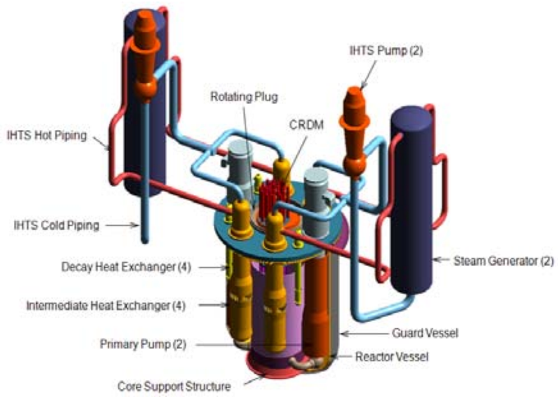
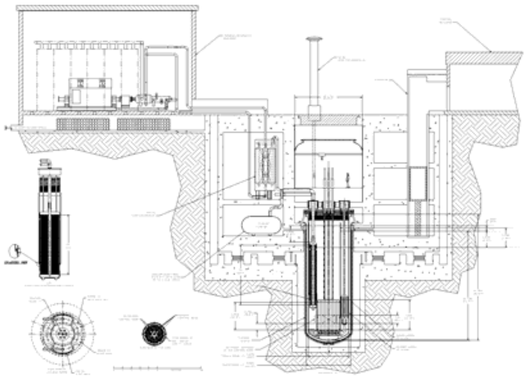


Figure 3: KALIMER (pool-configuration SFR)



The two primary fuel recycle technology options are (1) advanced aqueous and (2) pyrometallurgical processing. A variety of fuel options are being considered for the SFR, with mixed oxide the lead candidate for advanced aqueous recycle and mixed metal alloy the lead candidate for pyrometallurgical processing.

Figure 4: SMFR (small modular SFR configuration)



II. R&D ACTIVITIES

II.A. Status of co-operation

The System Arrangement (SA) for the international research and development of the SFR nuclear energy system became effective in 2006 and the present official members of the SA are:

- The French Alternative Energies and Atomic Energy Commission.
- The Department of Energy of the United States.
- The Joint Research Centre of Euratom.
- The Japan Atomic Energy Agency of Japan.
- The Ministry of Education, Science and Technology of the Republic of Korea.
- The China National Nuclear Corporation of the People's Republic of China.
- The State Atomic Energy Corporation "ROSATOM" of the Russian Federation.

Three Project Arrangements were signed in 2007 for Advanced Fuel, Component Design and Balance-of-Plant and Global Actinide Cycle International Demonstration. The Project Arrangement for Safety and Operation was signed in 2009. The Project Arrangement for System Integration and Arrangement is in a final stage awaiting a signing process.

II.B. R&D objectives

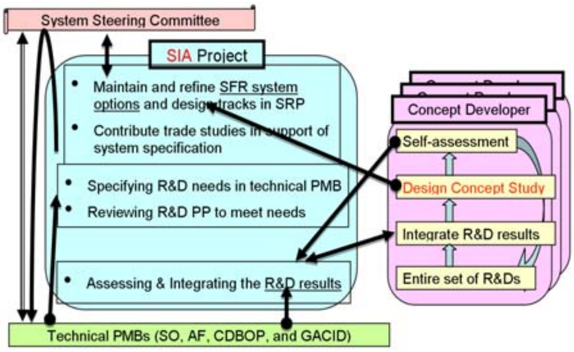
The SFR development approach builds on technologies already used for SFRs that have successfully been built and operated in France, Germany, India, Japan, the Russian Federation, the United Kingdom and the United States. As a benefit of these previous investments in technology, the majority of the R&D needs for the SFR are related to performance rather than viability of the system. Based on international SFR R&D plans, the research activities within GIF have been arranged by the SFR SA signatories into five projects. The scope and objectives of the R&D to be carried out in these five projects are summarized below.

II.B.1. System Integration and Assessment (SIA) project

GIF R&D collaborations are aimed at establishing the viability and enhancing the performance of the six selected Generation IV systems. While the ultimate benefit of this cooperative R&D will derive from future commercialization and deployment of the selected systems, the GIF collaborations themselves do not extend to the detailed design, demonstration, and deployment of these systems. Development of system *design concepts* to the point design, pre-conceptual or conceptual level is however an essential activity in GIF. This is needed to establish a consistent set of requirements for technology development, to ensure that the technologies developed are mutually compatible, and to allow the benefit of the R&D for system performance to be measured against the Generation IV Technology Goals.

The mission of the System Integration and Assessment (SIA) project is to carry out the integration and assessment functions for the Generation IV International Forum (GIF) sodium-cooled fast reactor (SFR). This role will help define and refine requirements for the overall SFR concept research and development (R&D), review and integrate results from the R&D projects to assure consistency, and periodically assess the system options and design tracks for conformance to Generation IV Technology Goals and other SFR-specific requirements. The activities of the SIA Project are carried out through a well defined, iterative process (see Figure 5).

Figure 5: Primary roles of SIA Project and relation to technical project



The SIA Project defines a comprehensive list of Generation IV SFR research and development needs. This effort is important to integrate the diverse activities of the technical R&D Projects to identify possible overlap or synergy opportunities.

The SIA Project will review and assess the results of the SFR technical Projects to establish guidance for the technical PMBs and recommendations for the SSC. This feedback will also consider the results of performance and safety studies of selected Generation IV SFR design tracks. These performance evaluations will rely on self-assessment results contributed by the Members.

A wide variety of interesting trade studies on the design and performance of SFR systems is being conducted by each of the System Arrangement member. These results may be particularly helpful for the integration of R&D contributions (e.g., evaluation of the performance impacts of key R&D innovations). Thus, member contributions of national study results that may be useful for concept design and integration are encouraged as SIA contributions.

II.B.2. Safety and Operation (SO) Project

In the safety area, the project involves R&D activities on phenomenological model development and experimental programs, conceptual studies in support of the design of safety provisions, preliminary assessment of safety systems, framework and methods for analysis of safety architecture. In the operation area, the project involves R&D activities on fast reactors safety tests and analysis of reactor operations, feedback from decommissioning, in-service inspection technique development, under-sodium viewing and sodium chemistry.

II.B.3. Advanced Fuel (AF) Project

Advanced Fuel Project aims at developing high burn-up MA bearing fuels as well as claddings and wrappers withstanding high neutron doses and temperatures. It includes: research on remote fuel fabrication techniques for fuels that contain minor actinides and possibly traces of fission products as well as performances under irradiation of fuels, claddings and wrappers. Candidates under consideration are: oxide, metal, nitride and carbide for fuels for homogeneous MA recycling, alternate fast reactor fuel forms and targets for heterogeneous MA recycling, and ferritic/martensitic and ODS steels for core materials.

II.B.4. Component Design and Balance-of-plant (CDBOP) Project

Research on Component Design and Balance-of-Plant has the objective of enhancing SFR system performance reducing the capital cost per unit electrical power and the cost of electricity generation. Primary research and development activities include advanced components and technologies to enhance the economic competitiveness of the plant, development of advanced in-service inspection instrumentation and repair methods using different approaches and technologies, research and development on advanced energy conversion systems such as the supercritical CO_2 Brayton cycle to improve plant economics and eliminate sodium-water reactions, and innovation in advanced, high reliability Rankine cycle steam generator designs and related instrumentation to enhance the robustness against sodium-water reaction as well as efficiency. In addition, the importance of the experience and lessons learned from the operation and upgrading of SFRs is recognized and summarized.

II.B.5. Global Actinide Cycle International Demonstration (GACID) Project

The project of "Global Actinide Cycle International Demonstration" (GACID) aims at conducting collaborative R&D activities with a view to demonstrate, on a significant scale, that fast neutron reactors can indeed manage the actinide inventory to satisfy the Generation IV criteria of safety, economy, sustainability and proliferation resistance and physical protection. The project consists of MA bearing test fuel fabrication, material properties measurements, irradiation behaviour modelling, Joyo irradiations, licensing and pin scale irradiations in Monju, and post-irradiation examination, as well as transportation of MA raw materials and MA bearing test fuels.

II.C. Milestones

The key milestones of the SFR system R&D projects are given below.

II.C.1. SIA Project

- Definition of SFR system options.
  - **2011**: Initial specification of SFR system options and design tracks.
- Definition of SFR R&D needs.
  - **2009**: Review and refine SFR R&D needs in the SRP.
- Review of assessments of SFR design tracks.
  - **2012**: Compile existing self-assessment results for SFR design tracks.
  - **2012**: Solicit economics assessment using ESWG methodology.
  - **2013**: Solicit proliferation assessment using PRPP methodology.
  - **2014**: Solicit safety assessment using RSWG methodology (note that review of safety performance self-assessment will likely be delegated to the SO Project).

II.C.2. SO Project

- Methods, models and codes.
  - **2008-2012**: Component and system models.
  - **2010-2015**: Transient and accident models.
  - **2012**: Codes and methodology validation.
- Experimental programs and operational experiences:
  - **2008-2012**: Basic phenomena studies.
  - **2010-2015**: Safety tests and analysis, tests for performance data.
  - **2008-2015**: SFR operational experiences.
- Studies of innovative design and safety systems:
  - **2008-2012**: Concepts of innovative safety provisions and systems.
  - **2010-2015**: Performance assessment of safety provisions and systems.
  - **2015**: Qualification of safety provisions and systems.

II.C.3. AF Project

- **2006-2015**: Preliminary evaluation of advanced fuels.
- **2008-2015**: Evaluation of MA-bearing fuels.
- **2008-2020**: High-burn-up fuel behaviour evaluation.
- **2021**: Demonstration and application of the selected advanced fuel.

II.C.4. CDBOP Project

- **2007-2012**: Viability study of proposed concepts.
  - Feasibility evaluation of proposed component concept.
- **2009-2015**: Performance tests for detailed design specification.
  - Selection of component concept.
- **2014-2016**: Demonstration of system performance.
  - Fixing the component design specification.

II.C.5. GACID Project

- **2007-2012**: Preparation for the limited MA-bearing fuel 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 MA-bearing fuel irradiation demonstration.
- **2013-2018**: [TBD] (Drafting for this period started in Nov. 2011).

III. MAIN ACTIVITIES AND OUTCOMES

III.A. System Integration and Assessment Project

To assist the integration of R&D activities, the SIA Project has identified several system options that define general classes of SFR design concepts: loop configuration, pool configuration, small modular reactor. Furthermore, within this structure several design tracks have been identified with pre-conceptual design contribution by SFR Members: JSFR (Japan), KALIMER (Korea), ESFR (Euratom), and SMFR (United States) as shown in Section 1.

Table 1 summarizes the key design parameters of the SFR design concepts. It is important to note that all of these SFR systems are designed with a large degree of flexibility in size, specific fuel design, and fuel loading configuration. These particular designs are indicative of current international SFR design studies that cover a wide range of power applications (sized from 50-1 500 MWe). With regard to the fuel and loading, any of the systems can be designed for different actinide management missions. The converter mode designs given in Table 1 could readily be modified to breeder or transmuter configurations by changing the fuel assembly design to impact the uranium loading. Furthermore, the SFR reactor performance can be achieved with different fuel forms, depending on the success of the advanced fuels research to develop and demonstrate recycle fuels.

Table 1: Key design parameters of generation IV SFR concepts

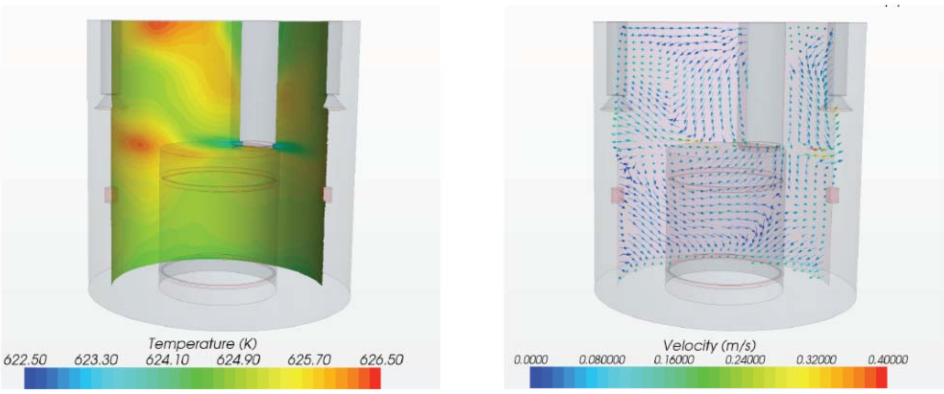
Design Parameters	JSFR	KALIMER	SMFR	ESFR
Power rating, MWe	1 500	600	50	1 512
Thermal power, MWt	3 570	1 500	125	3 600
Plant efficiency, %	42	40	~38	42
Core outlet coolant temperature, °C	550	545	~510	545
Core inlet coolant temperature, °C	395	390	~355	395
Main steam temperature, °C	503	503	480	490
Main steam pressure, MPa	16.7	16.5	20	18.5
Cycle length, years	1.5–2.2	1.1	30	1.35
Fuel reload batch, batches	4	5	1	5
Core diameter, m	5.1	4.2	1.75	4.72
Core height, m	1.0	0.89	1.0	1.0
Fuel type	MOX(TRU bearing)	Metal(U-TRU-10%Zr Alloy),	Metal(U-TRU-10%Zr Alloy),	MOX
Cladding material	ODS	HT9M	HT9	ODS
Pu enrichment (Pu/HM), %	13.8	25.2	15.0	15.7
Burn-up, GWd/t	150	139	~87	150 (max)
Breeding ratio	1.0–1.2	0.74	1.0	1.0–1.2

III.B. Safety and Operation Project

The development and validation of safety analysis codes is challenging issue in the process of new concept evaluations. The performance of system analysis codes such as CATHARE and MARS-LMR has been evaluated for the Phénix end-of-life test data. The applicability of the codes to SFR system was also investigated through the safety assessment of advanced SFR designs. The development of methods for uncertainty quantification and sensitivity assessment in advanced simulation techniques for design basis and safety analyses was continued. Example calculations using a prototype sampling technique applied to an unprotected loss-of-flow/loss-of-heat-sink accident analysis were performed. The integration of 3-D computational fluid dynamics models and system safety code SAS4A/SASSYS-1 was pursued to demonstrate the applicability for the modelling of multidimensional phenomena (Figure 6).

In regard to severe accidents, the applicability of severe accident codes (SAS-4A and SIMMER-III) to CDA sequence analyses was investigated and the dominant factors in the initiating phase and transition phase of unprotected events were identified. Experimental analyses were performed using SIMMER-III on fuel-pin disruption and low-energy distributed core motion, and the computer code models of SIMMER-III for severe accident analysis were improved.

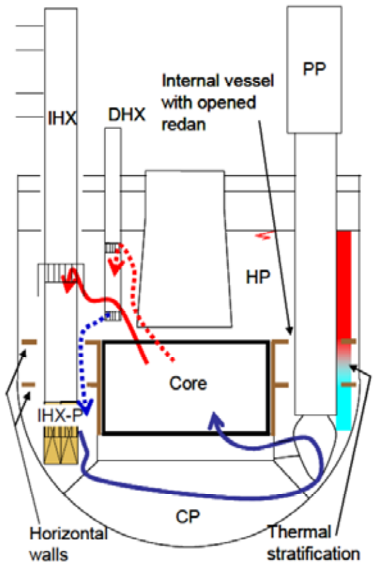
Figure 6: Temperature and velocity fields predicted by SAS4A/SASSYS-1 coupled with STAR-CCM+



With the final shutdown of the Phénix reactor plant, an ambitious program of End of Life tests has been set up, in order to complete and extend the data-base for code validation. Regarding to the thermal hydraulics topics, and especially system code, the major thermal hydraulics test is a test of Natural Convection. The test of Natural Convection performed in the frame of the Phénix End of Life Tests permits to extend the assessment of CATHARE system to natural convection transients. The detailed design of experimental facility was completed for the study on the evaluation of performance of a passive decay heat removal circuit. All the components of the test loop were manufactured and installed to compose the STELLA-1 facility. A review of international SFR testing experience to enhance safety performance characteristics and reduce safety margins in an advanced recycle reactor was performed, and a web-based database that contains all the available test data were developed. It was intended to evaluate ISI methodology through implementation of ISI for existing reactors and feedback on inspection technology associated to analysis of generic ISI situation, recommendation for SFR. The transportation of radioactive corrosion products and deposition behavior in the primary cooling circuit of MONJU was predicted using the PSYCHE code. Experience from Phénix operation and feedback was compiled and presented.

Innovative design concepts and provisions were investigated and their performances were assessed in order to evaluate whether the design meets the safety requirements. Feasibility studies of new vessel architecture for a pool type SFR concept, designated the stratified redan concept (Figure 7), was performed to improve hydraulic path of the natural circulation for decay heat removal and a better compactness. The effectiveness of design measures for the elimination of recriticality by the early discharge of molten oxide fuel and post-accident material redistribution was assessed based on the EAGLE experiment data (Figure 8).

Figure 7: Diagram of a stratified redan SFR

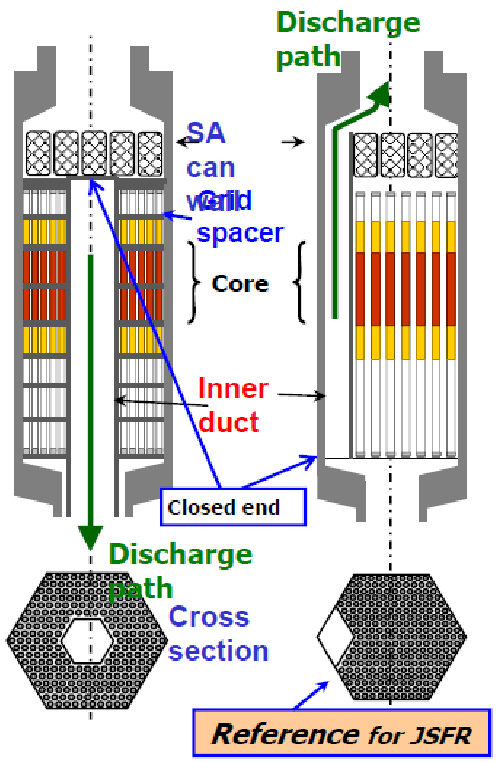


The materials to be used in core catchers for the case of a hypothetical core meltdown accident were also investigated. Systems analysis of the reactor shutdown system related to internal initiating events, and seismic response analysis considering characteristics of the advanced seismic isolation system were performed. The analyses of bounding design basis and ATWS transients for the conceptual design of an advanced recycle reactor, evaluating passive safety performance characteristics of oxide and metallic-fuelled core options were performed.

III.C. Advanced Fuel Project

Fuels under consideration are mixed uranium-plutonium based fuels: oxide, metal, nitride and carbide as SFR driver fuel with MA incorporation up to a few percent in accordance with the so-called homogeneous MA recycling in nuclear systems. A first technical evaluation based on historical experience, knowledge on fast reactor fuel development, as well as specific fuel tests currently being conducted on MA bearing fuels, has pointed out that both oxide and metal fuels emerge as primary options to meet quickly the goals. Regarding core materials, promising candidates are ferritic/martensitic steel and ODS (oxide dispersion steel). Fuel investigations have been enlarged since 2009 to include the heterogeneous route for MA transmutation, for which MA are concentrated in dedicated fuels located at the core periphery, by request of SIA project. Fabrication, irradiation and post-irradiation examinations have been performed regarding MA bearing metal, oxide, nitride and carbide fuels, and $(U,MA)O_{2-x}$ targets. Thermal properties of both MA-bearing driver fuels and $(U, MA)O_{2-x}$ have been measured. Developments on MA bearing fuel fabrication processes in hot cell by remote operation have continued. Regarding cladding development, cladding of Ferritic/Martensitic steel and ODS were fabricated and characterized. Preparation of fuel pins with ODS cladding for irradiation in Joyo have continued.

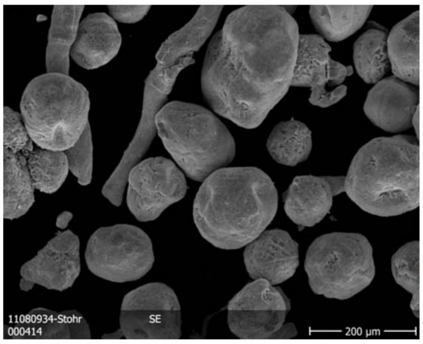
Figure 8: FAIDUS design for recriticality elimination



III.D. Component Design and Balance-of-plant Project

The CDBOP project started on 11 October 2007 when the Project Arrangement was signed by the members of CEA/France, DOE/USA, JAEA/Japan and KAERI/ROK. Euratom and the Russian Federation have shown interests in participating in the CDBOP PMB and presented their potential future technical contributions to the project from 2012. Euratom is expected to join the project as a new member and provide results on the definition of in-service inspection and repair requirements, and cycle optimization and component studies of AECS with $S\text{-}CO_2$.

Figure 9: MA bearing fuel and cladding tube fabrication



MA oxide fuel
$(U_{0.76},Pu_{0.16},Am_{0.04})O_{2-x}$

HT9 Cladding Tube

The CDBOP activities include lessons learned from SFR upgrading including ISIR, component and piping replacement, R&D on ultrasonic viewing technologies in sodium, high temperature LBB assessment for advanced component and piping materials, and S-CO_2 Brayton cycle development and demonstration including investigation of sodium-CO_2 interactions.

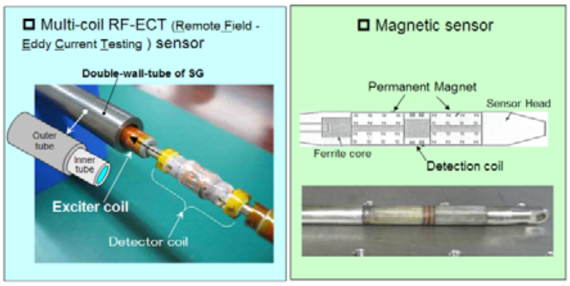
In-service inspection technologies, high temperature LBB assessment, S-CO_2 Brayton cycle AECS, and development of steam generators have been studied in 2009-2012. In the study of in-service inspection technologies [1][2], information was exchanged on the ongoing development of new and complementary in-service inspection technologies for in-vessel sodium components involving ultrasonic sensors both inside (under sodium viewing) and outside of the sodium, and of the new inspection technology for ISI of double-walled steam generator tubes. Two kinds of sensors have been developed to inspect the double-walled tubes of a Double-Walled Tube Steam Generator (DWT-SG); that is, a multi-coil Remote Field Eddy Current Testing sensor and a magnetic sensor (Figure 10). A 10 m long plate-type ultrasonic waveguide sensor, which was developed to overcome limitations of previous rod-type waveguide sensors and immersion sensors, was tested in sodium environments. The inside surface of the radiating end section of the 1.5 mm thick waveguide plate was coated with 0.25 mm thick beryllium (Be) to decrease the angle of the radiation beam and to generate a well-developed beam profile in sodium. The outer surface of the radiating end section was coated with 0.1 mm thick nickel (Ni) and micro-polished to obtain a surface roughness within 0.02 µm to enhance sodium wetting. A signal-to-noise ratio of 10 dB was achieved and characters spelling "SFR" with a 2 mm slit width were successfully recognized in sodium using a 10m long waveguide sensor (Figure 11).

Data on fatigue crack growth (FCG) and creep crack growth (CCG) for Mod.9Cr-1Mo Gr.91 ferritic steel specimens was reported. CCG tests of Gr.91 Heat Affected Zone metal were performed at 600°C and FCG tests of Gr.91 Compact Tension specimens and Single Edge Crack Tension Specimens were performed at 500°C, 550°C and 600°C for 0.1Hz and 1.0Hz loading frequencies, respectively, to be utilized in the high temperature LBB assessments.

Results have been contributed on the testing of a small-scale S-CO_2 compressor, testing of a split flow S-CO_2 compression loop, development of control strategies for S-CO_2 Brayton cycles, Argonne National Laboratory (ANL) Plant Dynamics Code calculations for a SFR incorporating a S-CO_2 Brayton cycle, experiments on sodium-CO_2 interactions, CO_2 corrosion and carburization tests, validation of modelling in the ANL Plant Dynamics Code through comparison with data from the Sandia National Laboratories S-CO_2 testing loop, and data on sodium plugging in compact diffusion-bonded heat exchanger sodium channels. A preliminary design of a SFR with a S-CO_2 cycle has also been developed to evaluate its dimensions [3].

Using the ANL Plant Dynamics Code, a new automatic control strategy has been developed for utilizing a S-CO_2 Brayton cycle with SFRs involving a new control mechanism of controlling the rotational speed of the shaft connecting the turbine and compressors when the cycle is disconnected from the electrical power grid [4]. The new strategy enables the cycle to be utilized for removing heat from the reactor not only at power generating conditions but also down to shutdown decay heat levels. This capability for the S-CO_2 cycle reduces the required capacity and cost of the SFR shutdown heat removal system further improving the benefits to SFR economics of utilizing this AECS concept.

Figure 10: Sensors for DWT-SG tube inspection

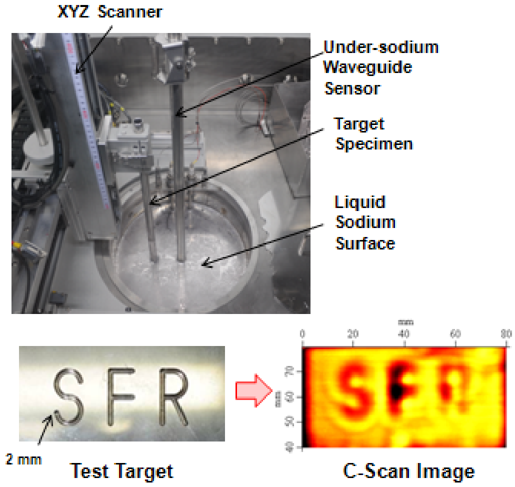


[This study is the result of "Technical development program on a commercialized FBR plant" entrusted to JAEA by the Ministry of Economy, Trade and Industry (METI).]

Starting in 2011, studies of sodium-heated steam generators (SGs) have been joined as a new theme of the project. The CDBOP members discuss design concepts, inspection technologies, sodium-water reaction phenomena, detectors and instrumentation, and thermal hydraulic properties, based on each nation's R&D results. As a development in SG design, 2D and 3D thermal hydraulic computer calculations of a DWT-SG were carried out. A modular SG approach to safety with respect to Sodium-Water Reaction (SWR) was studied to determine the benefits which could be gained from a robust safety demonstration.

The deliverables have been shared among the CDBOP members. Collaboration has been carried out between CEA and ANL since 2009 that includes applications of the ANL Plant Dynamics Code. Since 2010, this collaboration is now providing deliverables to the project. In 2013, an up to date technical paper on the works performed in the project shall be presented at FR13 [5].

Figure 11: Plate waveguide sensor test in sodium



III.E. Global Actinide Cycle International Demonstration Project

The Global Actinide Cycle International Demonstration Project (GACID) aims at demonstrating that the SFR can effectively manage all actinide elements – including uranium, plutonium, and minor actinides (MAs: neptunium, americium and curium) – by transmutation. The project includes fabrication and licensing of MA-bearing fuel, pin-scale irradiations, material property data preparation, irradiation behaviour modelling and post-irradiation examination, as well as transportation of MA raw materials and MA-bearing fuels. Bundle-scale demonstration will be included. This technical demonstration will be pursued using existing fast reactors in a reasonable time frame.

During 2011 the post irradiation examination of the minor actinide bearing fuel irradiated in the irradiation AM1 in the JOYO reactor has been achieved. All the results were reported during the annual meeting.

The irradiation AFC-2C and 2D have been performed by DOE in the ATR material testing reactor in Idaho. Preliminary irradiated fuel characterisations have been realized and presented to the GACID memberships.

R&D on fabrication is in progress and the specifications of (U, Pu, Am, Np) O_X have been established at CEA. The overall programme on properties measurements was defined and split between several laboratories.

IV. CONCLUSION

Over the past 50 years several experimental and prototype SFRs have been constructed and operated. Current GIF SFR R&D focuses on actinide fuel development for waste management, capital cost reduction features, and design features that promote safety with significant margins for bounding events.

Progress has been made in:

- Evaluation of advanced fuels such as metal, oxide, nitride, and carbide.
- Establishing feasibility of actinide recycling.
- Implementation of innovative safety.
- Improvement of high reliability in-service inspection and repair equipment.
- Development of S-CO_2 Brayton cycles.

ACKNOWLEDGEMENTS

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NOMENCLATURE

AF	Advanced Fuel
CDBOP	Component Design and Balance-Of-Plant
DOE	Department of Energy
GACID	Global Actinide Cycle International Demonstration
GIF	Generation IV International Forum
ODS	Oxide dispersion steel
PMB	Project Management Board
S-CO_2	Supercritical CO_2
SA	System Arrangement
SFR	Sodium-cooled Fast Reactor
SIA	System Integration and Assessment
SO	Safety and Operation

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VERY HIGH TEMPERATURE REACTOR (VHTR) SYSTEM

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I. ORIGINAL VHTR VISION

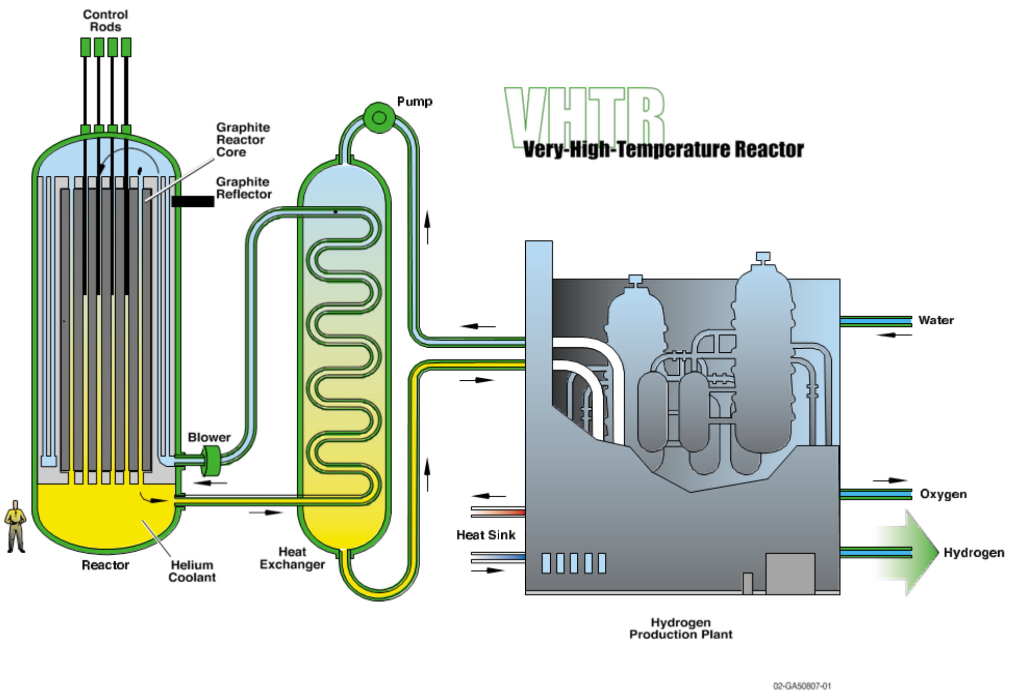
Among the six candidates of the Gen IV nuclear systems in the Technology Roadmap of the Generation IV International Forum (GIF), the Very High Temperature Reactor (VHTR) is primarily dedicated to the cogeneration of electricity and hydrogen, the latter being extracted from water using thermo-chemical, electro-chemical or hybrid processes. Its high outlet temperature makes it attractive also for the chemical, oil and iron industries. With an outlet temperature of 1 000°C, the VHTR allows the efficient production of hydrogen by thermo-chemical processes. Specific core layouts and robust components together with low power density enable passive decay heat removal. The VHTR has the potential for inherent safety, high thermal efficiency, process heat application capability, low operation and maintenance costs, and modular construction.

II. REFERENCE CONFIGURATIONS AND FEATURES

The VHTR is the next step in the evolutionary development of high-temperature gas-cooled reactors (HTGR). It uses TRISO coated particle fuel, graphite as moderator and reflector, helium as coolant and features a thermal neutron spectrum. It can supply nuclear heat and electricity over a range of core outlet temperatures between 700 and 950°C, and possibly more than 1 000°C in the future. The reactor core type of the VHTR can be a prismatic block core as in the Japanese HTTR, or a pebble-bed core as in the Chinese HTR-10. Although the shape of the fuel elements for the two configurations is different, the technical basis is the same: use of TRISO coated particle fuel in a graphite matrix, a fully ceramic (graphite) core structure, helium coolant and low power density to achieve high outlet temperatures and the retention of fission products inside the coated particle fuel under normal and accidental conditions. The VHTR can run on various fuel cycles such as U-Pu, Pu, MOX or U-Th.

For electricity generation, two power conversion options exist: either a helium gas turbine system directly placed in the primary coolant loop (direct power conversion cycle), or, at the lower end of the outlet temperature range, a steam generator feeding a conventional Rankine cycle. For nuclear heat applications such as process heat for refineries, petrochemistry, metallurgy, and hydrogen production, the heat application process is generally coupled to the reactor through an intermediate heat exchanger (IHX), in a so-called indirect cycle configuration. The VHTR can produce hydrogen using only heat and water through thermochemical processes such as the sulfur-iodine (S-I) process or the hybrid sulfur process, high temperature steam electrolysis (HTSE), or from heat, water, and natural gas by applying the steam reformer technology. The reference VHTR system that produces hydrogen is shown in Figure 1.

Figure 1: The VHTR for hydrogen production



A 600 MWth VHTR dedicated to hydrogen production can yield over 2 million normal cubic meters per day. The VHTR can also generate electricity with a high efficiency, ~50% at 950°C, compared with 47% at 850°C (in the case of a direct Brayton cycle). Co-generation of heat and power makes the VHTR an attractive heat source for large industrial complexes. The VHTR can be deployed near refineries and petrochemical plants to substitute large amounts of process heat (usually from gas firing) at different temperatures, including hydrogen generation for upgrading heavy and sour crude oil.

While the original approach for the VHTR at the start of GIF focused on very high outlet temperatures and hydrogen production, current market assessments have indicated that electricity production and industrial processes based on high temperature steam that require more modest outlet temperatures (700-850°C) have the greatest potential for application in the next decade and also reduce technical risk associated with higher outlet temperatures. As a result, over the past decade, the focus has moved from higher outlet temperature designs such as GT-MHR and PBMR to lower outlet temperature designs such as HTR-PM in China and the NGNP in the USA.

The high degree of safety of the HTGR/VHTR continues to be a strong motivation for coupling the system to industrial processes. Demonstrations of the safety performance for both the pebble and prismatic concepts at HTTR, HTR-10 and AVR have reinforced the value of the strong negative temperature coefficient, the high heat capacity of the graphite core, large temperature increase margin, and the robustness of TRISO fuel in producing a reactor concept that does not need off-site power to avoid radioactive release and to survive multiple failures or severe natural events such as those that occurred in the Fukushima accident.

III. KEY VHTR DEVELOPMENT TARGETS

There are still some technical challenges for the VHTR in the fields of fuel and materials, especially for the target of a core outlet temperature of 1 000°C.

Process-specific R&D gaps need to be filled to adapt the chemical process and the nuclear heat source to each other with regard to temperatures, power levels, and operational pressures. Heating of chemical reactors by helium is a departure from current industrial practice and needs specific R&D and demonstration. The development of intermediate heat exchangers, ducts, valves and associated heat transfer fluids are needed to provide process heat to many of the chemical processes.

The viability of producing hydrogen to support the process heat requirements also needs further study. Any contamination of the product will have to be avoided. Development of heat exchangers, coolant gas ducts, and valves will be necessary for isolation of the nuclear island from the production facilities. This is especially the case for isotopes like tritium, which can easily permeate metallic barriers at high temperatures.

In the past ten years, HTGR/VHTR R&D activities have been conducted in China, Japan, the United States, the European Union, Korea, France, South Africa, Russia, and other countries such as Kazakstan, Saudi Arabia etc. Signatories to the GIF VHTR System Arrangement up to the end of 2012 included Canada, China, Euratom, France, Japan, Korea, Switzerland, and the United States.

The signatories of the VHTR system agreed to cooperate on four research topics and formed the corresponding project management boards (PMBs) to support the future deployment of VHTR:

- Materials (MAT): development of high temperature alloys for vessels, IHX/SG, etc., development and qualification of high temperature graphite, development of ceramic composites for control rods, core internals and insulation.
- Fuel and Fuel Cycle (FFC): fuel fabrication, fuel qualification, fuel performance model; fuel cycle back-end (spent fuel and irradiated graphite).
- Hydrogen Production (HP): two thermo-chemical processes and high temperature electrolysis.
- Computational Methods Validation and Benchmarks (CMVB): code improvement for design and licensing, experiments for code validation.

IV. STATUS OF VHTR DEVELOPMENT

Over the past decade, significant advances have been made in the key technologies necessary to deploy a VHTR.

The GIF research activities are based on the national projects of the members. Some projects are specific to the GIF research plan, others are related to national research programs or to industrial projects.

In China, the 10MW high temperature test reactor (HTR-10) was built and has been operating since 2000. HTR-10 provides a good test bed for the TRISO coated particle fuel, the system and components, safety demonstrations, and code verification and validation for the VHTR, especially for the pebble bed concept. At the same time, a 200MWe pebble bed high temperature gas cooled reactor demonstration plant (HTR-PM) has been designed, and construction has started in 2012. HTR-PM is based on mature technologies, such as single zone pebble bed core, and steam cycle power conversion. HTR-PM contains two nuclear steam supply system modules (2 reactors plus 2 steam generators) and one common steam turbine. The key components and systems in HTR-PM will be tested at full scale. Most of the technology is based on the experience of HTR-10.

In Japan, HTTR reached criticality in 1998. The outlet temperature was raised to 950°C in 2004. 50 days of continuous operation at 950°C was completed in March 2010. Many safety experiments were carried out in HTTR, for example, in December 2010, a Loss of Forced Cooling accident at 30% power was successfully simulated. More safety tests are being conducted under the OECD-NEA LOFC Project which started in 2011. After HTTR, a 50MWt HTGR was designed for electricity generation and district heating, targeting developing countries. A new concept, named Naturally Safe HTGR (NSHTR) with improved inherent safety features was proposed and studied. Japan has also made significant progress in the development and demonstration of the I-S process for hydrogen production.

In the United States, the Next Generation Nuclear Plant (NGNP) project has the goal of demonstrating electricity, process heat and/or hydrogen production with the VHTR. The related R&D activities cover many fields such as fuel and graphite qualification, method development, and economic analyses. For hydrogen production, both IS and high temperature steam electrolysis process were tested and demonstrated.

In Korea, two projects were launched with hydrogen production as the main driver, in particular to reduce CO_2 emissions from steel making. For the R&D, the Nuclear Hydrogen Key Technologies Development Project (2006-2017) focuses on the development of key technologies for fuel, materials and high temperature experiments, computer codes and hydrogen production. The Nuclear Hydrogen Development and Demonstration (NHDD) Project has a longer term schedule up to 2026. The Industry Alliance for Nuclear Hydrogen has already been established and includes seven nuclear industrial companies or institutes, and five potential end-users.

In the European Union, several VHTR-related research projects cover the qualification of fuel and materials, and graphite fuel and graphite recycling, modeling and hydrogen production.

In France, although there is no dedicated VHTR research project, there is complementary work on the gas-cooled fast reactor which contributes to GIF projects on Hydrogen Production and Materials.

In Canada, complementary work performed for the SCWR also contributes to the GIF projects on Hydrogen Production and Materials.

Switzerland's contribution is mainly in the area of materials.

South Africa contributed to the Fuel and Fuel Cycle, Material, and Computational Methods Validation and Benchmark Project Arrangements up to the shutdown of the PBMR project in 2010.

From the viewpoint of Project Arrangements in the VHTR system, good collaboration among different member countries has been demonstrated, and solid technical progress has been achieved, especially in the projects MAT, FFC and HP.

On materials, significant achievements have been made in the following areas:

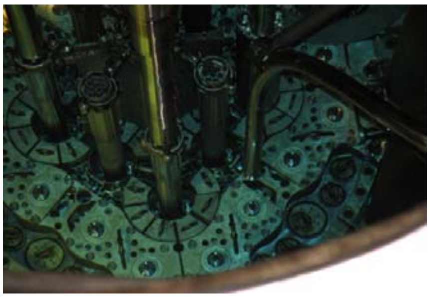
- Metallic materials, graphite, ceramic (composite) materials development.
- Irradiation testing in the USA, Switzerland, Euratom.
- Mechanical testing by all participants.
- Data consolidation in the ORNL Material Database.
- Development of design codes and standards.

Of course, there are still some R&D challenges in the fields of development of high temperature alloys, qualification of new graphite types, and development of composite ceramic materials. Some of the research on materials can benefit from and contribute to the research performed for other GIF systems.

On fuel, significant and effective collaboration was demonstrated:

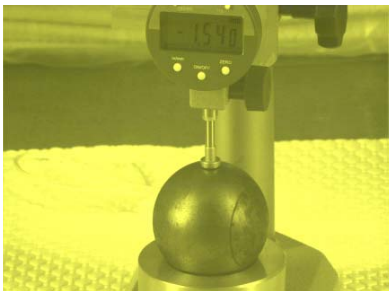
- Fuel irradiations in the ATR (US) (AGR irradiations, see Figure 2) with samples from the US, France and South Africa.
- Fuel irradiation project KR-HANARO in Korea, under preparation.
- Fuel irradiations in the HFR (EU) with fuel samples from EU and China.
- Coated particle irradiation in the HFR (EU) with samples from EU, France, Japan and Korea.

Figure 2: View into the ATR reactor pool used for fuel and material irradiations



For PIE and safety tests on irradiated fuel, results are available from the EU and the US (AGR-1 and HFR-EU1bis irradiations), and soon from HFR-EU1 (see Figure 3). The activities of the IAEA CRP6 also provide many valuable results concerning the fuel performance modeling.

Figure 3: Post-irradiation metrology of a fuel pebble



The EU CARBOWASTE project provides many results on waste management, which are made available to the GIF VHTR FFC members. The latter are also invited to an international CARBOWASTE workshop on this subject.

These achievements reinforce the confidence in the performance of current SiC based TRISO coated particle fuel. In addition, the development of new TRISO fuel with ZrC coating is on-going. In the field of fuel cycle, research concerns mainly the investigation of options for final disposal of spent fuel, and the recycling of irradiated fuel and graphite.

On hydrogen production, the main technology directions are the Iodine-Sulphur process, High Temperature Steam Electrolysis, and the Copper Chloride process. The main contributors are Japan, the United States, Korea, Canada, China and the EU. Lab-scale hydrogen production was achieved, but more work is required to integrate the processes for overall demonstration, to couple the process to the reactor, and to reduce the cost of of components, before scaling-up the system to commercial scale hydrogen production.

V. CURRENT VISION

In the original GIF Technology Roadmap, the target for the VHTR outlet temperature was set as 1 000°C, driven by the need for process heat at 950°C for the iodine-sulfur process for hydrogen production. The VHTR is the only concept out of the 6 GIF systems which is suitable for high efficiency hydrogen production. Obviously, higher outlet temperatures raise the technical challenges in terms of materials, fuel and components. Also, the delivery of heat at these temperatures is not trivial. However, studies in several countries have pointed at the large and growing market demand for high temperature steam for industrial process heat applications (for instance in the petrochemical and chemical industries). Other applications which could develop include coal liquefaction, desalination and of course, hydrogen production. Delivery of process heat from nuclear energy can help reduce CO_2 emissions and reduce the dependence on fossil fuel supply.

These non-electric applications are also accessible to the current modular HTGR designs with outlet temperatures between 700 and 950°C. This range is high enough for many applications. And the HTGR with outlet temperature of 700 to 950°C is a proven technology, demonstrated in experimental reactors (AVR, HTTR or HTR-10) and commercial scale demonstrations (THTR, Peach Bottom 1, Fort St. Vrain). With the concept of inherent safety of modular HTGR, and recent achievements in fuel, material, components and methods, modular HTGRs with outlet temperatures lower than the VHTR target of 1 000°C are ready for commercial deployment, such as the HTR-PM in China (whose construction has started) and the NGNP in the United States. Further work is required to facilitate licensing, standardize, reduce costs and develop coupling technologies for process heat applications.

Thus, the development of VHTR can be divided into two stages. The first stage is the HTGR with outlet temperature between 700 and 950°C, focusing on nuclear cogeneration of electricity and process heat, while the second stage is the VHTR itself with outlet temperature of 1 000°C, aimed at hydrogen production and other applications.

VI. CONCLUSION REMARKS

After 10 years of research and development, the original GIF goals set for the VHTR are recognized to be still largely valid: high safety level, high efficiency and capability to deliver process heat.

In terms of technology readiness, HTGRs with outlet temperatures between 700 and 950°C are ready for deployment. Examples of recent commercialization projects include the HTR-PM in China, now under construction and aimed at electricity production, and the NGNP in the United States.

The VHTR with outlet temperature of 1 000°C is a longer term research target for specific applications that require such high temperatures. These two stages of VHTR development and the will to cooperate in the related R&D activities are reflected in the four GIF VHTR projects, FFC, MA, HP and CMVB.

NOMENCLATURE

CMVB	Computational Methods Validation and Benchmarks (PMB)
FFC	Fuel and Fuel Cycle (PMB)
GIF	Generation IV International Forum
HP	Hydrogen Production (PMB)
HTGR	High Temperature Gas cooled Reactor
HTSE	High Temperature Steam Electrolysis
IHX	Intermediate Heat eXchanger
MAT	Material (PMB)
PMB	Project Management Board
VHTR	Very High Temperature Reactor

SUPERCRITICAL WATER COOLED REACTORS

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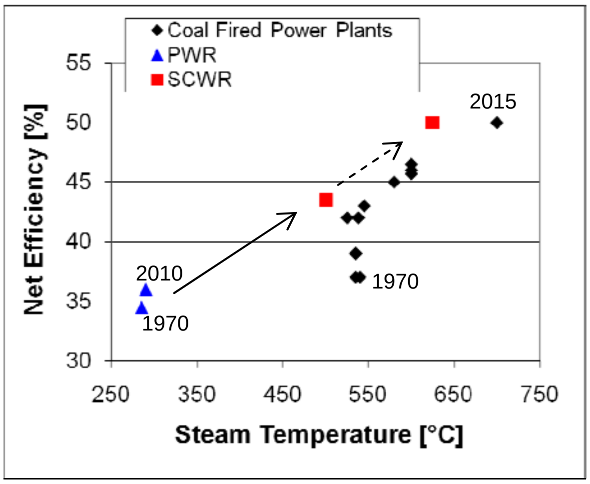
I. INTRODUCTION

The Supercritical Water-Cooled Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership, and pressure tube concepts proposed by Canada, generically called the Canadian-SCWR. Other than the specifics of the core design, these concepts have many similar features (e.g., outlet pressure and temperatures, steam cycle options, materials, etc.). Therefore, the R&D needs for each reactor type are common; this enables collaborative research to be pursued.

Figure 1: Future potential of water cooled reactors; a comparison with fossil fired power plants



As sketched in Figure 1, the idea of the SCWR follows the trend of coal fired power plants within the last 40 years, which succeeded to increase the net efficiency while decreasing the specific capital cost by increasing the live steam temperature and pressure. Since about 1990, these power plants have usually been designed for supercritical steam conditions, reaching up to 50% efficiency in the near future, whereas water-cooled reactors are still built with similar parameters as in 1970.

Similarly, the main advantage of the SCWR will be improved economics because of the higher thermodynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and proliferation resistance and physical protection are also possible and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

There are currently four projects for the SCWR System:

- System Integration and Assessment (SI&A).
- Thermal-Hydraulics and Safety (TH&S).
- Materials and Chemistry (M&C).
- Fuel Qualification Testing (FQT).

Table 1 lists the members and shows the status of arrangements signed for these projects. In addition, Russia signed the SCWR System Arrangement in 2011 and expressed its interest to join the projects. China is not yet a signatory of the SCWR System Arrangement and Project Arrangements but has been active in SCWR R&D as well. Their activities are not covered here.

Table 1: SCWR members and status of arrangements

Arrangement type	Signatories	Date of signature
System	CA, EU, J, RU	Nov. 2006 July 2011
Project SI&A	Managed by Steering Committee	
Project TH&S	CA, EU, J	Oct. 2009
Project M&C	CA, EU, J	Dec. 2010
Project FQT	Being negotiated	

In 2009, the members issued a System Research Plan for the SCWR [1], concentrating on the following five years with an outlook until around 2020. The following key priority R&D projects were identified:

- System integration and assessment: Definition of a reference design, based on the pressure tube and pressure vessel concepts, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance.
- Thermal-hydraulics and safety: Significant gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed. The design-basis accidents for a SCWR have some similarities with those of conventional water reactors, but the difference in thermal-hydraulic behaviour and large changes in fluid properties around the critical point compared to water at lower temperatures and pressures need to be better understood.
- Materials and chemistry: Selection of key materials for use in in-core and out-of-core components of both pressure tube and pressure vessel designs remains the key challenge. Selection of a reference water chemistry which minimizes materials degradation and corrosion product transport is being developed based on materials compatibility and an understanding of water radiolysis.
- Fuel qualification test: An important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.

Today, we can take a look at how many of these tasks have been completed up to 2012 and, consequently, what remains to be done. The following section gives an overview of the major achievements of each of these projects. Further detailed results on SCWR design and technology can be found e.g. in the proceedings of the 5th International Symposium on SCWR, held in March 2011 in Vancouver, with more than 140 paper submissions.

II. SCWR SYSTEM INTEGRATION AND ASSESSMENT (SI&A)

In general, the SCWR can be designed as a pressure vessel type reactor, cooled and moderated with light water, or as a pressure tube type reactor moderated with heavy water. In a pressure vessel design, the neutron spectrum can be thermal, for which additional water channels will be needed in the reactor core, or the spectrum can be fast if these water channels are omitted. Common to all design options is a supercritical pressure of more than 22.1 MPa, a superheated core outlet temperature of 500°C or more and a once-through steam cycle such that the produced steam is directly supplied to the high pressure turbine and any coolant recirculation in the primary system is omitted.

The System Research Plan [1] has been following all these options in different member states to build up a suitable decision basis for further design and development. According to the System Research Plan, the basic design phase should be completed by the end of 2012 with an assessment with respect to the Generation IV criteria. Collaboration with the economic modeling working group and with the working group for proliferation resistance and physical protection has been helpful to compare results with other GIF systems.

Design and construction of a prototype or demonstration unit has not yet been foreseen in the System Research Plan but is planned to be included in its next update.

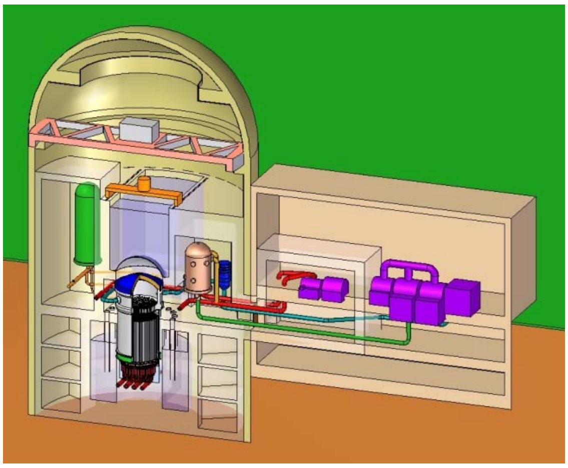
II.A. Canadian SI&A activities

The concept of the Canadian-SCWR features a thermal neutron spectrum core and maintains the CANDU modular design with fuel channels separating the light water coolant from the heavy water moderator.

Operating conditions are set at a pressure of 25 MPa and at a core outlet coolant temperature of 625°C, matching closely the advanced high-pressure turbine designs currently implemented in SCW fossil power plants. The high outlet temperature would lead to an estimated thermal efficiency of 48%. Further enhancement of the thermal efficiency can be achieved with the reheat option and co-generation capability. The thermal power of the concept can be varied to meet user requirements and has been set at 2 540 MW for reference purposes. This results in an electric power output of about 1 200 MW. The supercritical steam is led directly to the high-pressure turbine, eliminating the need for steam generators (plant simplification and cost saving). Figure 2 illustrates the pre-conceptual plant layout of the Canadian SCWR.

Different from the conventional CANDU design, the pressure tubes are vertical to avoid thermal stratification and on-line refueling has been given up to reduce the diameter of the end flanges. Moreover, the use of thorium fuel is considered as an attractive option. Design activities are still continuing.

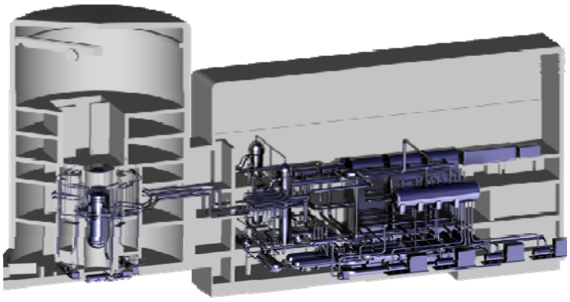
Figure 2: Layout of the pre-conceptual Canadian-SCWR



II.B. European SI&A activities

In Europe, a conceptual design, called the High Performance Light Water Reactor (HPLWR), has been completed and assessed with respect to the criteria of the Generation IV International Forum. This SCWR concept features a thermal reactor core with a thermal power of 2300 MW, a core outlet temperature of 500°C and a supercritical core inlet pressure of 25 MPa. Key design features are a coolant heat-up in three steps with intermediate coolant mixing to minimize the peak cladding temperature below 650°C. The compact containment design with only 25m inner height includes automatic depressurization systems, a pressure suppression pool, as well as four redundant high and low pressure coolant injection systems with residual heat removal. The once through steam cycle produces a net electric power of 1 000 MW, resulting in a net efficiency of 43.5%. Figure 3 illustrates the plant layout with its reactor and turbine building.

Figure 3: Layout of the HPLWR nuclear power plant with 1 000 MW net electric power



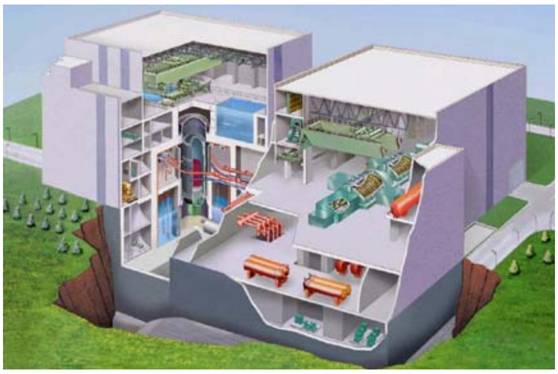
The basic design phase was completed in 2010 and the overall assessment of this concept, summarized by Starflinger et al. [2], confirmed that the reactor core will meet the design criteria of maximum cladding temperatures and maximum fuel temperatures at design conditions, even including uncertainties and allowances for operation. Safety analyses performed do not give any indication that the core melting frequency could be higher than in current LWRs due to the intrinsic characteristics of this concept. The expected economic advantages are obvious in the plant erection costs, being around 20% cheaper than current LWR, whereas the expected fuel costs do not show significant cost savings yet. Assessment of the proliferation resistance shows at least no disadvantage compared with current LWR. The thermal design concept is not considered for a sustainable fuel cycle, but recycled MOX fuel could be used optionally. A recent textbook, edited by Schulenberg and

Starflinger [3], summarizes the design details and analyses of this concept.

II.C. Japanese SI&A activities

In Japan, the concept of the Japanese Supercritical Water Cooled Reactor (JSCWR) has been developed and was assessed in 2010 with respect to the criteria of the Generation IV International Forum. The electrical output is assumed to range from 600 MW to 1 700 MW to fulfill user's requirements, and a reference value of 1 620 MW has been selected. The main charac-teristics of this concept are a thermal neutron spectrum using light water as moderator and coolant, a pressure-vessel type core, a once through reactor, and a direct Rankine cycle turbine system. Figure 4 shows a bird's eye view of the power plant. Detailed core design and safety system analyses confirm the viability of the concept. The expected economic advantages agree reasonably well with those of the European plant concept mentioned above, and the proliferation resistance is expected to be as good.

Figure 4: Bird's eye view of the JSCWR power plant



Further design studies on the reactor core and on safety systems and technologies needed for the SCWR have been performed by the University of Tokyo. A textbook on Super Light Water Reactors and Super Fast Reactors has been issued by Oka et al. [5], summarizing their results. Their thermal core option features a heat up in two steps, designed as a downward flow in the first stage and an upward flow in the central part of the core as a second step, and includes intermediate coolant mixing underneath the core. The concept has a core outlet temperature of more than 500°C. The fast core design option is based on a similar flow path. A heterogeneous arrangement of seed and blanket assemblies ensures a negative void coefficient during the entire burn-up cycle.

III. SCWR THERMAL-HYDRAULICS AND SAFETY (TH&S)

Supercritical water is a single phase fluid having liquid-like properties below the pseudo-critical temperature (384°C at 25 MPa) and steam-like properties above this temperature. Heat transfer of supercritical water differs fundamentally from ordinary fluids in the vicinity of the pseudo-critical point, where the fluid properties vary significantly with temperature. Heat transfer in this range can be enhanced at low heat flux compared with ordinary fluids, or dete-riorated at high heat flux and low mass flux, causing local hot spots on the heated surface. Prediction of such hot spots still remains a challenge. Up to now, simple heat transfer correlations cannot predict these phenomena properly and computational fluid dynamics (CFD) or even large eddy simulations are taken instead. Similar questions arise with critical flows through orifices or breaks and with stability limits of supercritical fluids in heat exchangers if the pseudo-critical point is located in the computational domain. New physical models and codes describing these phenomena need to be validated by experiments with supercritical water or at least with surrogate fluids having similar properties, like supercritical CO_2 or refrigerants.

A second part of this project covers innovative concepts of safety systems for SCWRs and their transient analyses with system codes. These include loss of coolant accidents, loss of power accidents, loss of flow accidents, and other scenarios which may cause a risk for the power plant and should be assessed conceptually, accompanying the design work. Collaboration with the Risk and Safety Working Group enables a comparison with other GIF systems.

According to the SCWR System Research Plan, the time frame until end of 2012 was used for heat transfer tests with supercritical water and with other, surrogate fluids in tubes, annuli and bundles as well as for stability and critical flow tests. Integral tests of the envisaged safety systems are foreseen for a later phase. Today, suitable concepts of active safety systems are available for SCWRs, but passive safety systems and their experimental validation still remain to be a challenge.

III.A. Canadian TH&S activities

Canada has been focusing on establishing infrastructure for thermal-hydraulics research. A number of test facilities have been designed and constructed in Canada. These facilities are established mainly for heat-transfer tests with tubes, annuli, and bundle subassemblies in water, carbon dioxide, or refrigerant flows. At this point, the design of the water-test facility is complete and construction has been initiated. A refrigerant and a carbon dioxide test facilities have been constructed for supercritical heat-transfer experiments co-vering test sections including tubes, annuli, a 3-rod bundle, a 4-rod bundle, and a 7-rod bundle. Figure 5 shows a view of the carbon-dioxide test facility [6]. Axial surface-temperature distributions were obtained with 8 mm and 22 mm tubes.

Figure 5: Upper view of the heat transfer test facility with carbon dioxide flow

At the subcritical pressure of 6.7 MPa (i.e., lower than the critical pressure of 7.38 MPa for carbon dioxide), nucleate boiling is observed for length-to-diameter ratios up to about 180 at a mass flux of 451 kg/(m^{2}s) and 270 at the mass flux of 1 476 kg/(m^{2}s) (see Figure 6). Departure from nucleate boiling occurred at these locations, beyond which film boiling is observed.

At supercritical pressures of about 9 MPa, deterioration heat transfer is observed with a sharp rise in surface temperature at a mass flux of 425kg/(m^{2}s) and a heat flux of 42 kW/m^2 (length-to-diameter ratio of about 160) (see Figure 7). Another surface-temperature peak observed at the length-to-diameter ratio of about 190 corresponds to the pseudo-critical temperature. Deterioration heat transfer is not observed at other test conditions in the figure.

Figure 6: Wall temperature measurements at sub-critical pressures in an 8 mm tube with carbon dioxide flow

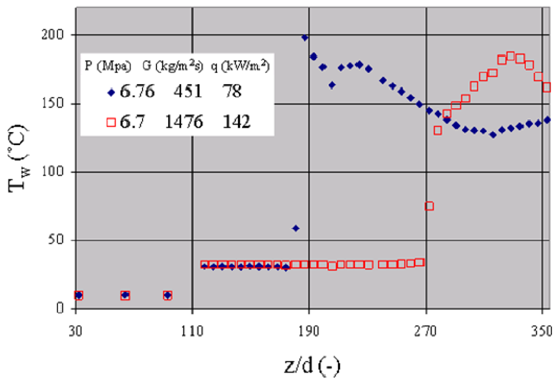
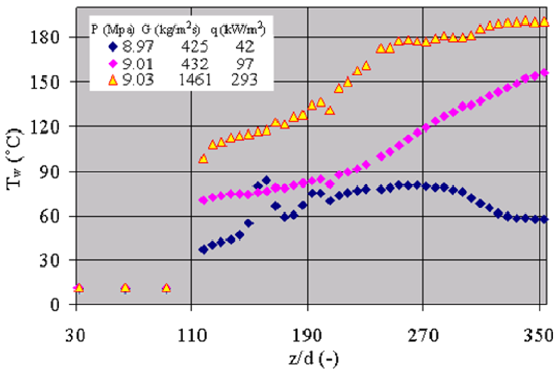
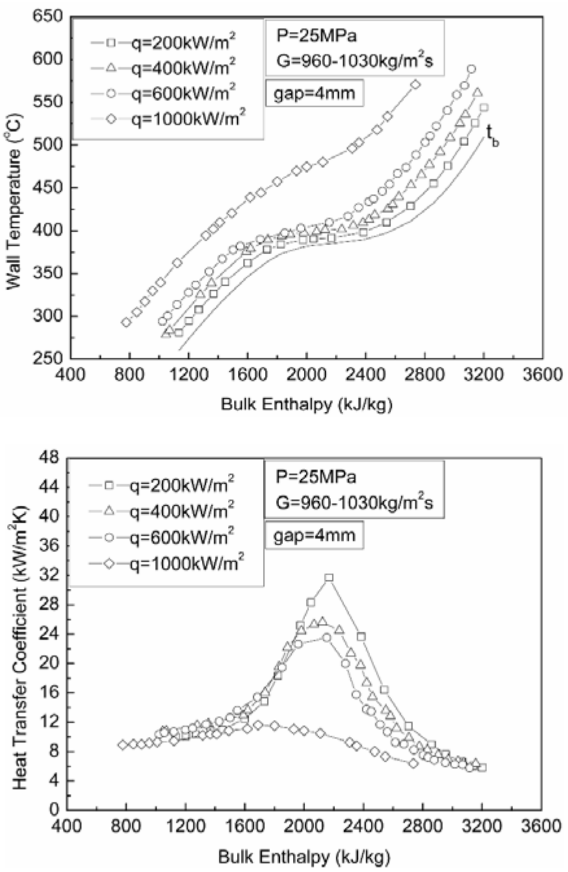


Figure 7: Wall-temperature measurements at supercritical pressures in an 8 mm tube with carbon dioxide flow



A heat-transfer experiment has been completed with annuli of two different flow areas in supercritical water flow [7]. The inner heater element has an outer diameter of 8 mm, while two different outer unheated flow tubes with inside diameters of 16 mm (i.e., 4 mm gap size between inner and outer tubes) and 20 mm (i.e., 6 mm gap size) have been used. The test section was installed vertically in the loop and tested with an upward flow of supercritical water. Inlet and outlet fluid temperatures, outlet pressure, and pressure drop over the test section were measured. Wall temperature measurements have been obtained over a range of mass fluxes and heat fluxes at outlet pressures of 23, 25, and 28 MPa. Figure 8 illustrates variations of wall temperature, and corresponding heat-transfer coefficient, with local enthalpy and heat flux. Deteriorated heat transfer has been observed at a heat flux of 1 000 kW/m^2.

Figure 8: Wall temperature measurements obtained from the super-critical water heat-transfer test with an annulus



Surface-temperature measurements were also obtained for the assessment of effects of gap size (or flow area) and spacers on heat transfer in annuli. Enhanced heat transfer in the annular test section was shown with the 6 mm gap size, compared to 4 mm gap size, at similar local conditions and heat flux. The difference is larger at low heat flux and high mass flux conditions than at high heat flux and low mass flux conditions. The effect of spacers is strong on heat transfer. Heat-transfer coefficients at the location of the spacer are consistently larger than those at locations further away of the spacer.

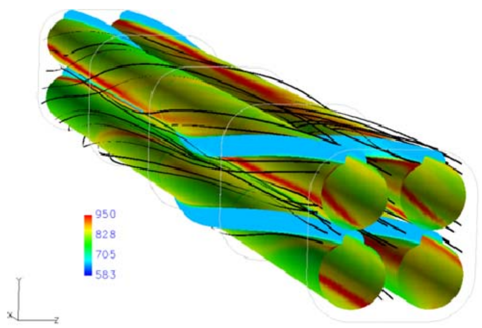
The supercritical heat-transfer database has been expanded to include water and carbon dioxide data previously obtained at the University of Manchester. These data cover mainly the mixed-convection region and are applicable for model development and validation.

A look-up table for heat-transfer coefficients covering subcritical and supercritical conditions has been developed. It covers two film-boiling regions (i.e. inverted annular flow and dispersed flow) at subcritical pressures and three regions (i.e. liquid-like, gas-like and pseudo-critical) at supercritical pressures.

III.B. European TH&S activities

The European consortium has been working on predictions of heat transfer. An example of a numerical prediction of heat transfer in a rod bundle is shown in Figure 9. These CFD analyses of Chandra et al. [8] show the surface temperature of fuel rods at a heat flux of 1 375 kW/m^2 and a mass flux of 1 332 kg/m^{2}s, predicted with FLUENT. The bulk temperature is 310°C at the inlet. A wire is wrapped around the fuel rods to serve a grid spacer and as a coolant mixing device. The analysis shows that the wire is also improving the local heat transfer.

Figure 9: Predicted surface temperature in K on fuel rods with wire wrap [8]



A safety concept has been developed and analyzed for the HPLWR, as summarized by Schulenberg and Starflinger [3]. The main safety functions and appropriate strategies for accident control have been identified, and the key parameters for the operation of the systems have been selected. The transient analyses performed address a variety of initiating events, including anticipated transients as well as accidents. The analyses show that the safety systems can effectively limit overheating of the core under the most severe conditions, such as loss of coolant accidents and loss of flow transients.

A Dutch program on the stability of a natural-circulation driven SCWR has been finalized in 2010 by the Delft University of Technology [9]. The project encompassed both numerical work and experiments. It has been found that the stability shows similarities with a natural-circulation driven boiling water system, but one major difference is that it is possible to follow a trajectory from zero power conditions to nominal conditions without crossing an unstable region. The origin of this finding is the gradual change of the density of the supercritical water with respect to the temperature.

For future collaboration in TH&S, two programs in the field of turbulent heat transfer in supercritical flows have been initiated. One program, called THINS, started in 2010 and includes a specific work package on non-unity Prandtl number, turbulent flows [11]. The second program is of Dutch origin. Local measurements will be taken with Laser-Doppler Anemometry and Particle Image Velocimetry to validate the models.

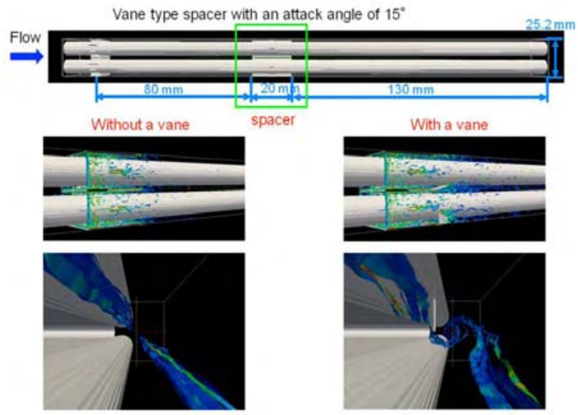
III.C. Japanese TH&S activities

In Japan, the development of the best estimate correlations on heat transfer and pressure drop was continued based on technical papers published by foreign researchers. Moreover, development of a thermal-hydraulic analysis method for thermal design of a SCWR was considered.

As for development of the thermal-hydraulic analysis method, consideration of the heat transfer augmentation due to spacers settled on the outer surface of fuel rods was performed. An effect of the heat transfer augmentation due to a spacer was taken into consideration in order to reduce the maximum fuel cladding surface temperature (MCST) from the current core design value. The target of a MCST decrease is 30-50 K. Spacer shapes were also considered to enhance the heat transfer coefficients.

As an example, the turbulent flows in the fuel assembly with a vane type spacer were predicted using a computational fluid dynamic tool, as can be seen in Figure 10. Figure 10 (a) shows a case without any vane and Figure 10 (b) shows a case with a vane.

Figure 10: Generation of large turbulence structures around fuel rods due to the vane on a spacer



(b) Without any vane (a) With a vane

Each numerical domain contains 2x2 fuel rods. The fuel rod diameter is 9.5 mm, the gap width between adjacent fuel rods is 3.1 mm, and the hydraulic diameter is 11.7 mm. The number of computational grids is 192x 192x640. Figure 10 shows the results under the conditions of supercritical water without any heat flux. Unsteady vortex structures are observed behind the spacer. By generating a large swirl flow due to the vane on a spacer, it was clarified quantitatively that turbulent intensities are strengthened.

III.D. Mutual benchmark study

The TH&S Project Management Board is organizing an international benchmarking exercise against supercritical water data obtained with a heated 7-rod bundle assembly. These data were obtained at Japan Atomic Energy Agency (JAEA) and have been submitted to the Project Management Board (PMB) as part of Japan's contribution. Facility data and experimental conditions were provided beforehand, and the experimental data afterwards. The benchmarking results will be presented during a workshop in September 2013 in Delft, the Netherlands.

This benchmark consists of a well-defined 7-rod bundle flow with grid spacers. The facility is equipped with a large number of thermocouples on the outer wall of the heating rods. Care has been taken to the azimuthal symmetry of the rod internals, ensuring a uniform distribution of power (or heat flux).

IV. SCWR MATERIALS AND CHEMISTRY (M&C)

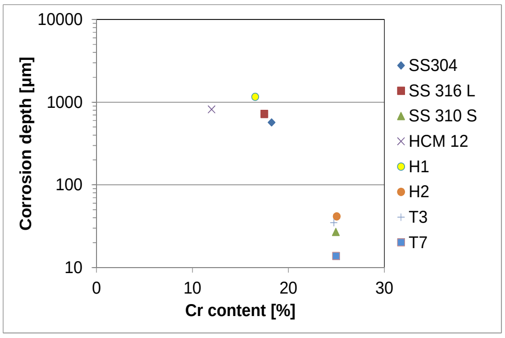
The identification of appropriate materials for in-core and out-core components is one of the major challenges for the development of the SCWR. For any SCWR core design, materials for reactor internals and fuel cladding need to be evaluated and qualified. Zirconium-based alloys, so pervasive in conventional water-cooled reactors, do not appear to be viable fuel cladding materials given the high peak cladding temperatures of the SCWR concepts. Based on the available data for other alloy classes, there is no single alloy that currently has received enough study to unequivocally ensure its performance in an SCWR. Although considerable experience is available for fast reactors and SCW-cooled FFPs, there is little or no data on the behavior of these materials inside an SCWR at the temperature and pressure of interest.

Another key component of this program is to develop an enhanced understanding of the chemistry of supercritical water. The marked change in the density of supercritical water through the critical point is accompanied by dramatic changes in chemical properties. These complications are further exacerbated by in-core radiolysis, which on-going studies show is markedly different from simplistic extrapolations of the behavior encountered in conventional water-cooled reactors.

Up to 2012, a large number of corrosion and stress corrosion cracking (SCC) tests have been conducted by all participants. The current database includes information on the general corrosion of more than 90 alloys. Tests have been conducted in a variety of facilities including static autoclaves, flow loops and pressurized capsules; a number of new test facilities for this purpose have been commissioned in the last four years.

Data on general corrosion for a range of candidate materials (including ferritic-martensitic steels, austenitic stainless steels, ODS steels and titanium) show a general trend of increasing corrosion resistance with increasing Cr content, as sketched in Figure 11. While stainless steels with less than 20% Cr are expected to fail as the corrosion depth would exceed the wall thickness, a Cr-content of around 25% has a potential to meet the design requirements of less than 10% corrosion depth within 50 000 h at 700°C. The beneficial effect of high Cr content in an SCWR is contingent upon being able to control water radiolysis, as the Cr oxides responsible for the corrosion resistance become soluble under very oxidizing conditions.

Figure 11: Predicted corrosion depth after 50 000 h at 700°C in supercritical water



Weight change measurements and a variety of metallographic and surface analysis techniques have been used to characterize high Cr materials following exposure to supercritical water. A significant finding was the rediscovery of the large effect of surface microstructure on corrosion for austenitic steels, e.g., polished versus machined. If the relevant mechanisms can be better understood, this may provide a means of imparting a higher corrosion resistance to materials currently not considered ideal candidates for use in an SCWR.

A major activity of the M&C PMB has been the organization of a round robin corrosion test program to compare the results of corrosion tests in different test facilities using a standard test protocol and coupon preparation method. The tests are now underway and will be completed in 2013.

IV.A. Canadian M&C activities

The core outlet temperature of the Canadian SCWR concept is higher than those of the EU and Japanese designs, and this presents a major challenge for material development. A significant amount of new infrastructure required for SCWR materials testing (autoclaves, corrosion and SCC test loops, creep apparatus) as well as facilities for production of oxide dispersion-strengthened alloys has now been established in Canada. A key activity has been the on-going development of a corrosion database to capture experimental data generated by Canadian R&D projects and GIF collaborations, as well compiled from open literature. A large parametric study of the effects of temperature, pressure, water chemistry and surface finish on corrosion was completed. Work is on-going to better understand corrosion mechanisms in SCW and to perform fundamental studies of oxidation resistance and corrosion mechanisms using model binary and ternary alloys, and molecular dynamics simulations of the structure of supercritical water at surfaces. The use of ceramic or metallic coatings to improve corrosion resistance of key components continues, including ceramic coating of P91 and zirconium alloys, and testing of NiCrAl(Y) and similar materials in supercritical water for up to 5 000 hours. A key enabling technology is the insulator required for the insulated fuel channel concept, and a major program is underway to develop and test candidate ceramic materials.

The specification of a chemistry control strategy is a major focus of the Canadian program, in particular the understanding of water radiolysis and corrosion product transport. In Canada, both experimental and modeling approaches are being used to develop an improved understanding of water radiolysis in SCW. The existing Monte Carlo model has been benchmarked against a recently released state-of-the-art assessment of all existing sub-critical water radiolysis data. Molecular dynamics simulations were carried out at different densities and temperatures to obtain a detailed picture of the heterogeneous molecular structure of SCW, needed to determine how this structure influences radiation energy deposition and subsequent radiolysis reactions. Work is on-going to determine the solubility of relevant metal oxides (e.g., magnetite, molybdenum oxides) and predict the corrosion product deposition. Initial studies of a model fission product (strontium) showed that neutral species are important at moderate concentrations at 350°C; their solubility in SCW is sufficient to allow transport out-of-core.

IV.B. Japanese M&C activities

Results from a detailed study on the corrosion of commercial SUS310S austenitic stainless steel and three other experimental alloys proposed by Hitachi (H2) and Toshiba (T3 and T7), shown in Figure 11, are among the most promising candidates. While there is no overall consensus on the best material for fuel cladding yet, there was general agreement that the Hitachi H2 modified 310 stainless steel containing Zr is the best candidate to be used as the reference material for the fuel qualification testing. However, more test data are needed at temperatures up to 700 °C for material qualification; the required tests could e.g. be performed in the VTT autoclave, which can reach up to 695 °C, or in high temperature, low pressure steam, which was recently shown to be a good surrogate for supercritical water above about 550 °C [12].

IV.C. European M&C activities

In the frame of European FP7 projects, SCC and general corrosion tests have been performed on ODS and austenitic steels to determine their corrosion resistances in SCW. In addition, environmental effect on creep rate has been studied on selected austenitic candidate alloys.

The higher material temperature, irradiation, and the thin walls of core components such as moderator box and fuel claddings are a combination of requirements which are more difficult to fulfil than in supercritical coal fired power plants. In this case, choices will be mainly high-performance stainless steels or novel oxide dispersion strengthened (ODS) steels. Other core components operating at 500°C may apply commonly used austenitic stainless steels. Ni-based alloys are excluded in core components because nickel has high neutron absorption cross-section and hence high Ni content adversely affects core neutronics. For thin-walled components especially, such as fuel cladding in the SCWR design, corrosion, stress corrosion cracking (SCC), and creep resistance are among the severest degradation modes needing to be understood and controlled.

Based on the results, the applicability of low Cr (< ~17 weight-%) austenitic and ODS steels is limited to temperatures less than 550°C. At 550°C, the oxidation rate increases rapidly for both alloy groups. In terms of general corrosion resistance, increasing Cr contents, e.g., 20% Cr, could extend the maximum operating temperature to 650°C or even higher. Creep test results, however, indicated that thin walled components made of austenitic stainless steels are prone to environmentally enhanced creep. This phenomenon requires further study. Cold working of the austenitic stainless steel (17-18% Cr) surface appears to suppress oxidation significantly up to 650°C for a substantial exposure time (at least up to 3 000 h). However, in SCWR, the exposure times are much longer.

In-pile tests of radiolysis and its effect on the water chemistry and corrosion are being prepared at Research Centre Rež (CVR). A supercritical water loop with an active channel inside the LVR-15 reactor has been constructed and commissioned in an out-of-pile test installation. It is ready for in-pile testing and will be installed inside the research reactor LVR-15 as soon as the required construction work in the reactor building is completed. The auxiliary unit with heaters and coolers, the purification system, water chemistry monitoring and sampling, and the dosing system are shown in Figure 12. Details of this system and recent results have been described by Ruzickova et al. [13]. Compared with the schedule of the System Research Plan, these important in-pile material and chemistry tests have had some delays to improve shielding of the test facility.

Figure 12: Auxiliary unit for in-pile radiolysis and water chemistry tests with supercritical water at UJV Rež, Ruzickova et al. [13]

During 2011 and 2012, the behaviour of austenitic steels 316L and 08CH18N10T (AISI 321) and ferritic-martensitic steels used for evaporator and boiler components in super-critical water cooled fossil fuelled power plants (P91, P92, Super304H, HR3C) were studied. The test conditions were just above the critical point (400°C, 25 MPa) to simulate evaporator conditions of the HPLWR. The loop is currently being used out-of-pile and the experiments performed are adapted for supercritical water cooled fossil fuelled power plants. The facility is now completely functional and experiments to support development of new equipment (e.g., specimen holders for specific measurements such as mechanical stress, etc.) for SCWR research are being performed (Figures 13 and 14). Specimen holders can be equipped with three point bending tests and interfaces for special sensors connections.

Figure 13: Specimen holder before exposure in the loop

Figure 14: 316L and 08CH18N10T specimens after exposure in the Czech supercritical water loop

The experiments performed in 2012 focused on the influence of water chemistry on the corrosion behaviour. The experiments were performed in pure water and using oxygen-ammonia water chemistry (Table 2).

The results are currently being evaluated using techniques such as SEM, gravimetry, ESCA, Mott-Schottky plots, etc. In 2013 the experiments will continue at the temperature of 600°C and pressure of 25 MPa.

Table 2: Water chemistry of the experiments performed in the Czech supercritical water loop

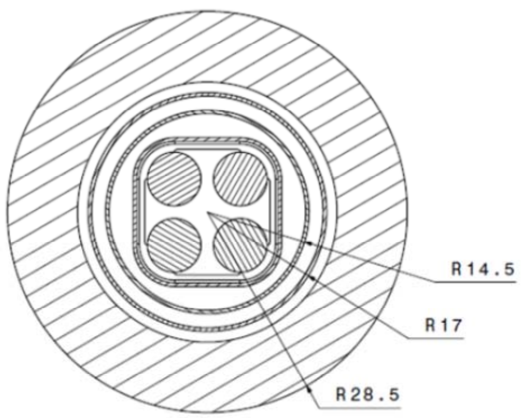
Parameter	Typical value	Maximum
pH	9.2 – 9.5	9.2 – 9.8
Oxygen	< 5 µg/l	10 µg/l
SiO_2	< 5 µg/l	20 µg/l
Fe	< 5 µg/l	10 µg/l
Na	< 2 µg/l	10 µg/l
Cu	< 3 µg/l	5 µg/l
TOC	< 100 µg/l	200 µg/l

V. SCWR FUEL QUALIFICATION TEST (FQT)

An in-pile test of a small scale fuel assembly, characterizing core design features of the SCWR is the subject of a new project, called the fuel qualification test, being negotiated between Europe and Canada. This test is planned to be the first application of supercritical water as coolant in a nuclear facility. Therefore, the design and licensing phase will be helpful to identify general problems expected during the licensing procedure for an SCWR. The tests will validate design codes like thermal-hydraulic predictions, neutronic and system code predictions as well as stress and deformation analyses and shall qualify the cladding material under reactor conditions. Qualification of the fuel rod manufacturing process and of monitoring systems for SCWR conditions are among the most challenging tasks to be performed before a prototype reactor can be built.

Four fuel rods with 8 mm outer diameter and with a wire wrapped around each rod as mixing spacer, like in the HPLWR core concept, are planned to be installed in a pressure tube of 57 mm outer diameter to replace an ordinary fuel assembly of the LVR-15 research reactor in Řež, Czech Republic. The four fuel rods, shown in Figure 15, will contain UO_2 pellets with an enrichment of less than 20%, providing a power of more than 63 kW over an active length of 60 cm. The maximum linear heat rate of 38 kW/m is close to the HPLWR design limit.

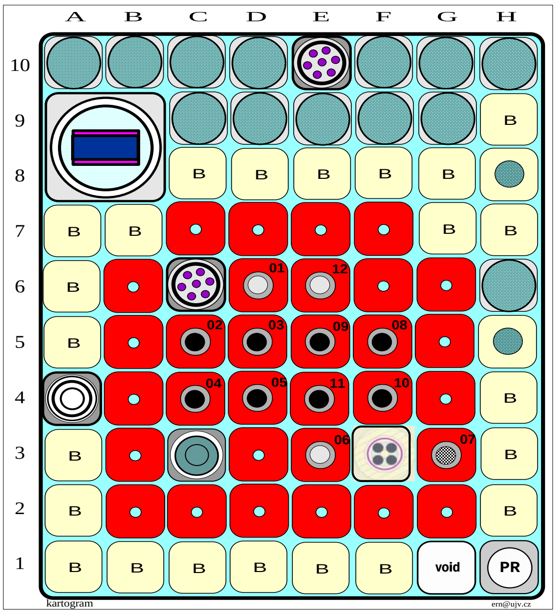
Figure 15: Cross section of the active test section of the in-pile fuel qualification test



The average linear heat rate is 26.5 kW/m. Supercritical water with 25 MPa pressure will entering the pressure tube at an inlet temperature of 300°C. It is first driven by the outer guide tube along the tube wall, keeping its peak temperature below 400°C, and then heated in a recuperator and by the gamma power released in the structural material to around 370°C before entering the test section. Before leaving the pressure tube, a U-tube cooler in the upper part of the pressure tube reduces the coolant temperature back to 300°C.

These test conditions represent the most challenging part of the HPLWR evaporator in which the bulk temperature of the coolant is slightly below the pseudo-critical temperature of 384°C at 25 MPa, but the cladding temperature is higher than the pseudo-critical temperature, such that a deterioration of heat transfer is challenged. The peak linear heat rate of the fuel rods corresponds to a peak heat flux of 1 500 kW/m², and the design mass flow corresponds to a coolant mass flux of 1 380 kg/m²s. For the first test series to be performed, the available stainless steel 316L, qualified for reactor applications, will be used for the fuel cladding, which implies that the peak cladding temperature must be kept below 550°C under normal operating conditions.

Figure 16: Cross section of the LVR-15 research reactor with potential core position of the pressure tube



The cross section of the LVR-15 research reactor, sketched in Figure 14, shows one potential core position of the pressure tube. The coolant loop outside the pressure tube includes a recirculation pump, a coolant make-up system and a sampling system, running at around 300°C. A bladder type accumulator, partly filled will with nitrogen, will help keep the system pressure stable. All these components are placed outside the reactor building. Inlet and outlet lines of the primary system will run inside a shielded duct through the reactor building to the primary block in a hall adjacent to the reactor hall. Details of the loop and its safety systems are discussed by Schulenberg et al. [14].

Design and assessment of the system is planned to be completed by end of 2013 and construction work is planned to start by the end of 2015. A Chinese consortium is currently supporting the design phase with their SCRIPT project, in which an out-of-pile test of the small fuel assembly shall validate the thermal-hydraulic and system codes used for design.

The proposed project plan includes initial tests up to 400°C coolant temperature with qualified cladding alloys to commission the test facility, followed by tests with elevated coolant temperatures up to 500°C using advanced high Cr stainless steels for the fuel claddings.

VI. OUTLOOK

So far, the SCWR research and development program has followed the System Research Plan defined in 2009, with only minor delay. Today, we have several design concepts which could serve as a basis for a prototype design, and a few more might still follow. The thermal-hydraulics of supercritical water are well understood, in principle, and potential material candidates have been identified. What needs to be done next?

The next step towards an SCWR prototype goes along with validation of thermal-hydraulic models and qualification of codes for which at least small scale component tests are needed, and with validation of innovative safety systems, requiring larger, integral tests, in particular if passive safety systems shall be included. New facilities for such thermal-hydraulic tests have just been built e.g. in Canada or in China, using supercritical water or surrogate fluids. Qualification of cladding alloys or other structural materials for supercritical water conditions will require more than just autoclave tests. E.g. the new in-pile supercritical water loop will provide more realistic conditions, close to those which are expected in an SCWR. A milestone within the next 10 years will be the in-pile tests of a fuel assembly under supercritical water conditions, for which materials and codes must be qualified, an effort which is similar as for a prototype test.

Realistically, a prototype can only be designed after experience has been gained with in-pile tests of single fuel assemblies. Therefore, different from other Generation IV concepts which had already been built similarly in the past, the SCWR System Research Plan did not specify a target date yet for a prototype. Early design studies, however, could easily be performed before these test results will be available.

It would be reasonable, therefore, to update the SCWR System Research Plan. Today, a number of new partners like research institutes in Russia or in China, who expressed serious interest in joining the system, could offer more opportunities than before. Their support could even over-compensate the temporary reduction of resources in Japan, caused by new R&D tasks due to the Fukushima accident.

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GAS FAST REACTOR SYSTEM

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ABSTRACT

The GFR system is a high-temperature helium-cooled fast-spectrum reactor with a closed fuel cycle. It combines the advantages of fast-spectrum systems for long-term sustainability of uranium resources and waste minimisation (through fuel multiple reprocessing and fission of long-lived actinides), with those of high-temperature systems (high thermal cycle efficiency and industrial use of the generated heat, for hydrogen production for example).

The reference concept for GFR is a 2 400 MWth plant operating with a core outlet temperature of 850°C enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economics in a fast reactor core. This represents the biggest challenge in the development of the GFR system. Significant progress has been made in establishing a workable concept for the fuel element since the last GIF symposium. In particular the reference concept has shifted from that of plate fuel to a more conventional pin bundle configuration. The second significant challenge for GFR is ensuring decay heat removal in all anticipated operational and fault conditions.

A necessary step in the development of a commercial GFR is the establishment of an experimental demonstrator reactor for the qualification of the refractory fuel elements and for a full scale demonstration of the GFR-specific safety systems. This demonstrator is ALLEGRO, a 75 MWth reactor with the ability to operate with different core configurations starting from a "conventional" core featuring steel-clad MOX-fuelled pins through to the GFR all-ceramic fuel elements in the latter stages of operation. A consortium of central European research organisations has become established to progress the design and to address the licensing issues with the intention of constructing ALLGERO as a research facility in central Europe.

I. INTRODUCTION

Out of the six energy systems covered under GIF (Generation IV International Forum), three concern purely fast neutron reactors (cooled with sodium, lead or gas), and the fourth one is the thermal neutron very high temperature reactor. Their specificities are summarised in Table 1. The two remaining systems have quite different characteristics from the former four.

The GFR system is a high-temperature helium-cooled fast-spectrum reactor with a closed fuel cycle. It combines the advantages of fast-spectrum systems for long-term sustainability of uranium resources and waste minimisation (through fuel multiple reprocessing and fission of long-lived actinides), with those of high-temperature systems (high thermal cycle efficiency and industrial use of the generated heat, for hydrogen production for example) [1, 2]. The GFR system arrangement (SA) was signed at the end of 2006 by the Generation IV International Forum (GIF) members Euratom, France, Japan and Switzerland. In addition to their national programmes, France and Switzerland are very active members within Euratom, with a number of organisations in France, and PSI in

Switzerland being members of the GoFastR project (Euratom FP7), which provides the main contribution from Euratom to the GIF GFR system development [3].

Two projects were discussed at the origin of the SA, dealing with conceptual design & safety (CD&S), and fuel and core materials (FCM). The conceptual design & safety project arrangement was signed in 2009 by Euratom, France and Switzerland, and is effective as of 17 December 2009. The Fuel and other core materials project arrangement remains unsigned and the participants have agreed to continue their collaboration on an informal basis.

II. MAIN CHARACTERISTICS OF THE GAS FAST REACTOR SYSTEM (GFR)

The reference design for GFR is based around a 2 400 MWth reactor core contained within a steel pressure vessel. The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hex-tube. The favoured material at the moment for the pin clad and hex-tubes is silicon carbide fibre reinforced silicon carbide. Figure 1 shows the reactor core located within its fabricated steel pressure vessel surrounded by main heat exchangers and decay heat removal loops. The whole of the primary circuit is contained within a secondary pressure boundary, the guard containment. The coolant is helium and the core outlet temperature will be of the order of 850°C. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle (Figure 2) containing a helium-nitrogen mixture which, in turn drives a closed cycle gas turbine.

The waste heat from the gas turbine exhaust is used to raise steam in a steam generator which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plant so represents an established technology, with the only difference in the GFR case being the use of a closed cycle gas-turbine.

The proposed experimental reactor ALLEGRO (formerly ETDR) could become the first gas-cooled fast reactor to be constructed. Being a small experimental reactor (75 MW_{th}), the objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as the fuel, the fuel elements and specific safety systems in particular, the decay heat removal function, together with demonstrating that these features can be integrated successfully into a representative system. So far, ALLEGRO development has been driven by the French national programme with significant contributions from Euratom and Switzerland. In 2010 a memorandum of understanding was signed between the Czech Republic, the Slovak Republic and Hungary as partners to support each other in bidding for one of them to host ALLEGRO, with assurances that the two other partners would provide technical and administrative support to the successful host nation.

Figure 1: GFR reference design

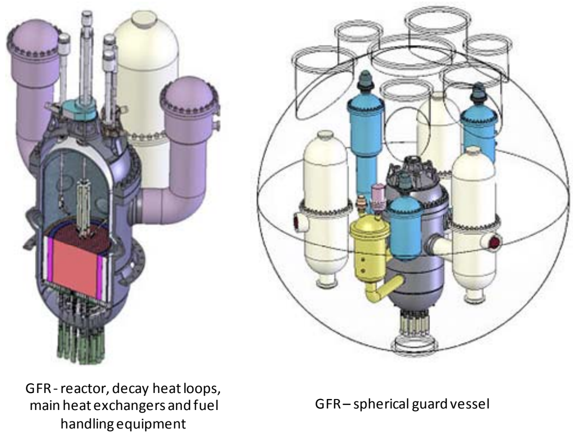
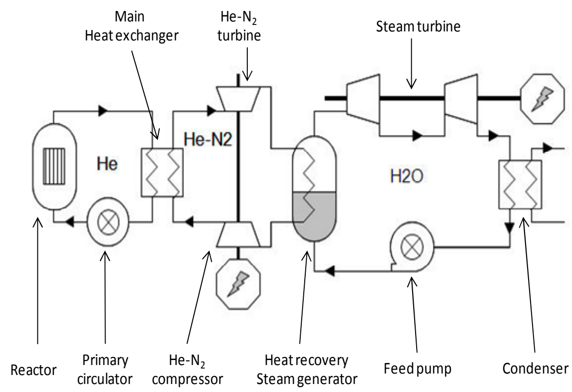


Figure 2: GFR indirect combined cycle power conversion system



III. MAIN R&D OBJECTIVES

The GFR uses the same fuel recycling processes as the SFR and the same reactor technology as the VHTR. Therefore, its development approach is to rely, in so far as feasible, on technologies developed for the VHTR for structures, materials, components and power conversion system. Nevertheless, it calls for specific R&D beyond the current and foreseen work on the VHTR system, mainly on core design and safety approach. These technology gaps are specific to GFR and must be addressed to demonstrate the technical (and commercial) viability of the reactor:

- Fuel forms suitable for simultaneous high-temperature and high power density operation with tolerance of fault conditions.
- Development of core materials with superior resistance to fast-neutron fluence under very high-temperature conditions with good structural, ageing and fission product retention capabilities.
- Core design, achieving a core that is self-sustaining in fissile material but, preferably, without the use of heterogeneous fertile "breeder" blankets to increase proliferation resistance and with the capability to burn minor actinides to improve sustainability.
- Safety systems, including highly reliable decay heat removal systems [4] that must cope with high core power density and the lack of any significant thermal inertia in the core or the coolant provided by the moderator in thermal reactor designs or the liquid metal coolant in other fast reactor systems.
- Fuel cycle technology, including spent-fuel treatment and refabrication for recycling uranium, plutonium and minor actinides.

In this context, the main goals of the conceptual design & safety (CD&S) project are:

- Definition of a GFR reference conceptual design and operating parameters (meeting requirements, already presented in previous reports, on breeding, MA transmutation, Pu mass, efficiency, availability and safety objectives).
- Identification and study of alternative design features (e.g. lower temperatures, pre-stressed concrete pressure vessel, diverse decay heat removal systems).
- Definition of appropriate safety architecture for the reference GFR system and its alternatives.
- Definition of the ALLEGRO conceptual design and its safety architecture, in coherence with that of the GFR.
- Development and validation of computational tools needed to analyse performance and operating transients (design basis accidents and beyond).

The goals of the fuel and other core materials (FCM) project are to investigate fuel element design and qualification, material for cladding, and dense fuel material:

- Regarding fuel design, with at least 50% of fissile phase inside the fuel element, pin-type fuel has been finally selected to enhance high power density.
- For clad, standard alloys cannot reach the foreseen temperature. Refractory materials have to be envisaged (metals and ceramic composite), while ODS alloy can be applied for lower temperature GFR core concepts.
- For achieving a high power density and a high temperature, dense fuels with good thermal conductivity are required. Carbide and nitride appear more attractive than oxide. However, oxide is a backup because of extensive experience feedback.

For the development of this innovative fuel element, the R&D activities performed within the FCM project include fuel element design, in-core materials studies (clad materials and fissile phase), fuel fabrication and irradiation program.

IV. MAIN ACTIVITIES AND OUTLOOK

GFR core design

CEA (France) has produced a design for a first 2 400 MW_{th} self-sustainable core with carbide pins and SiC cladding. This core forms the basis of all of the current system and transient analysis studies. Studies focusing on the fuel concept are still underway though the main trends are understood.

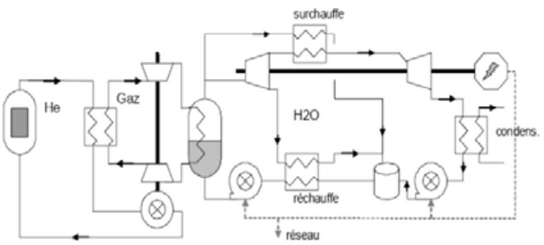
In conjunction with CEA, PSI has produced two documents that characterise the reference core using PSI's tools. The second of these documents is proposed to be considered as a complete neutronic specification of the GFR core.

GFR system design

The power conversion system of the GFR reference system design is an indirect cycle with helium on the primary circuit, a Brayton cycle with a mixture of nitrogen and helium on the secondary circuit and a steam cycle on the tertiary circuit. In particular, the primary compressors are driven by electrical motors.

Among alternative system designs studied, the "coupled cycle" option (CEA patent) appears particularly attractive. In this design, the primary circuit exchanges thermal and mechanical energy with the secondary one: the primary compressors are driven by the secondary turbomachine, i.e., the shafts connecting the turbines and the compressors of the secondary circuits are also connected to the corresponding primary blowers (Figure 3), via longer shafts crossing the primary circuit vessel. The secondary circuit and the tertiary circuit remains conceptually the same as the reference, except the mixture of nitrogen and helium, which is replaced by pure helium.

Figure 3: Principle scheme of the indirect coupled cycle: the primary blower is mechanically coupled to the secondary turbomachine



This option includes numerous assets, with at first the advantage to eliminate by design some of the loss of flow accidents which is particularly interesting for GFR safety demonstration. The attractiveness in terms of passiveness and autonomy is important: the main loops, by their natural adaptations to the primary thermal-hydraulic conditions, could be valued for long term core cooling either for pressurised or depressurised situation.

GFR safety systems design

AREVA (France, Euratom) has produced a deliverable that is to be delivered to the GIF entitled "Contribution to a report on review of technologies for DHR components". The DHR system of the GoFastR project is defined in continuity with the previous European FP6 GCFR project DHR system. Overall DHR strategy adopted during FP6 seems applicable. The report reviews the DHR main components, the valves, heat exchangers and gas blowers.

Scenario studies

NNL (UK, Euratom) has produced a scoping document and a final report on GFR penetration in a nuclear park. The study has used the ORION fuel cycle modelling code to analyse three fuel cycle scenarios:

- An all-PWR reactor fleet with a power output of 14 046 MWe (6 AP1000s and 5 EPRs).
- As (i) but with the addition of five GFRs phased in gradually while the PWRs are being phased out followed by a 7-year period where these five GFRs are allowed to become self-sustaining. An additional two GFRs are then introduced fuelled by the remaining PWR-sourced Pu.
- As (i) but with the addition of seven GFRs phased in ~30 years after the PWRs are closed down.

Whichever of the latter two options is chosen, this work demonstrates that GFRs can be integrated into an existing modern PWR fleet, with the Pu for the initial GFR (U,Pu)C fuel charge coming from reprocessed PWR fuel. The results also show that GFRs could be used to lower the amount of minor actinides in a fuel cycle. The fuel manufacturing requirements for typical operating scenarios have been quantified and the decay heats and radio-toxicities of the spent fuel determined.

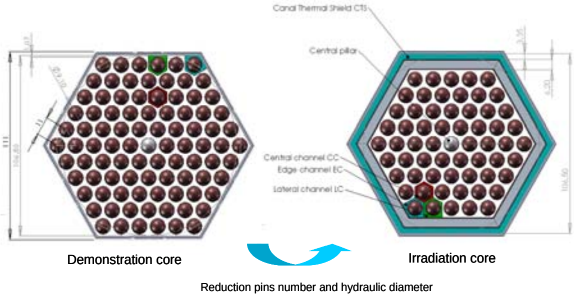
ALLEGRO core studies

The CEA has produced a first report as an entry point for all of the Euratom partners. This report also includes a proposal for the design of experimental GFR sub-assemblies to be loaded in the MOX starting core of ALLEGRO. Figure 4 illustrates such a sub-assembly in which the conventional steel wrapper tube is protected by a thermal barrier (on the right). Once these sub-assemblies are tested successfully, it will be possible to proceed to a whole core with GFR sub-assembly technology (on the left).

SRS (Italy, Euratom) has been working on a GFR-type sub-asssembly concept for the ALLEGRO demonstration core based on the idea of an hexagonal tube made of SiC plates held together within a metallic skeleton made of collars at different levels connected together with tie rods (Figure 5).

A high-temperature resistant alloy would be needed for the tie rods but they could be cooled by a helium bypass if necessary. The collars could also have a function of contact pads between adjacent sub-assemblies. Such a hexagonal tube could be used for the MOX feeding core and the experimental GFR sub-assembly, thus allowing a progressive transition from the MOX core to a full GFR technology core. More detailed studies are planned to be performed with realistic material properties.

Figure 4: Illustration of precursor GFR sub-assemblies to be tested in ALLEGRO

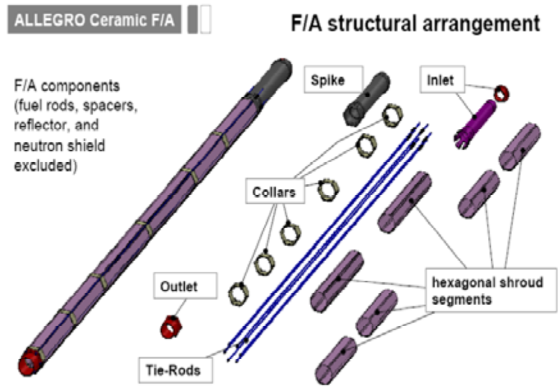


GFR transient analysis

The work on development of the computer models of the GFR system has started using the following system codes: TRACE/FRED (PSI), CATHARE (CEA/AEKI), RELAP5 (ENEA) and RELAP3D (ANSALDO).

The results of these analyses will be used to improve the design of ALLEGRO's safety architecture and the resulting cooling strategy continues to be developed in the frame of the Euratom GoFastR project.

Figure 5: Principal of composite (SiC/ metal) hexagonal tube



V. CONCLUSION

It is to be noted that, while France and Japan have been very active in the development of the GFR concept, providing remarkable results regarding conceptual design, safety assessment and fuel development in the previous years, in 2010 French research priorities were re-focused on sodium-cooled fast reactors, which led to a reduction of effort on the GFR system. Further, the Fukushima Daiichi accident in 2011 further refocused priorities away from GFR in Japan, and to a lesser extent in Switzerland.

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THE LEAD-COOLED FAST REACTOR PAST ACTIVITIES, PRESENT STATUS AND FUTURE DIRECTIONS

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ABSTRACT

The Lead-cooled Fast Reactor (LFR) features a fast neutron spectrum, high temperature operation, and cooling by molten lead or lead-bismuth eutectic (LBE), low-pressure, chemically inert liquids with very good thermodynamic properties. It would have multiple applications including production of electricity, hydrogen and process heat. System concepts represented in plans of the Generation IV International Forum (GIF) System Research Plan (SRP) are based on Europe's ELFR lead-cooled system, Russia's BREST-OD-300 and the SSTAR system concept designed in the United States.

The LFR has excellent materials management capabilities since it operates in the fast-neutron spectrum and uses a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner to consume actinides from spent LWR fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of molten lead as a chemically inert and low-pressure coolant. In terms of sustainability, lead is abundant and hence available, even in case of deployment of a large number of reactors. More importantly, as with other fast systems, fuel sustainability is greatly enhanced by the conversion capabilities of the LFR fuel cycle. Because they incorporate a liquid coolant with a very high margin to boiling and benign interaction with air or water, LFR concepts offer substantial potential in terms of safety, design simplification, proliferation resistance and the resulting economic performance. An important factor is the potential for benign end state to severe accidents.

The LFR has development needs in the areas of fuels, materials performance, and corrosion control. During the next five years progress is expected on materials, system design, and operating parameters. Significant test and demonstration activities are underway and planned during this time frame.

I. INTRODUCTION

This paper reviews the past history of development of the LFR, summarizes the current status, and presents on-going plans for future development. Past experience includes design development activities in several regions of the world as well as the significant deployment of a LFR technology in the Soviet Union for military (submarine propulsion) purposes. At present, the technical work underway by GIF participants includes activities associated with three different variants of the LFR representing three different systems sizes: the European Lead-cooled Fast Reactor (ELFR, 600 MWe); the Russian BREST-OD-300 (300 MWe); and the Small Secure Transportable Autonomous Reactor (SSTAR, 20 MWe) system concept designed in the US. Future activities include a variety of on-going and planned efforts to address remaining technical issues while proceeding toward demonstration of modern LFR concepts.

In this paper, we present an overview and historical backdrop of LFR development, the present status of GIF-LFR-PSSC, a summary of three reference LFR systems, a discussion of the advantages and challenges facing LFR development, and a summary of some considerations related to the safety attributes of LFRs under severe accident conditions in light of the Fukushima event.

II. THE HISTORICAL BACKDROP OF LFR DEVELOPMENT

The idea of fast reactors cooled by heavy liquid metals is an unfamiliar one to many, yet there is a considerable past history related to such reactors. In fact, the first fast reactor to operate was Clementine, a fast spectrum reactor cooled by the heavy liquid metal mercury. Clementine operated from 1946 to 1952 with a maximum output of 25 kWt. [1] After that time, operating experience with heavy liquid metal cooled reactors shifted to the Soviet Union/Russia while more recent design and experimental work has been carried out relatively broadly throughout the world. In 2002 the GIF identified the LFR as one of the six promising nuclear energy technologies to be considered for future advanced systems. [2]

II.A. Russian Experience with LBE- and lead-cooled reactors

Significant industrial and operational experience with reactors cooled by lead or lead-bismuth-eutectic (LBE) was gained by Russia (the Soviet Union and then the Russian Federation) in their program to design, produce and deploy LBE-cooled reactors for submarine propulsion during the period from the mid 1960s until the 1990s. During this period, a total of 12 reactors and 15 reactor cores were built and deployed, including two that were operated onshore. In total, this program represented about 80 reactor-years of operating experience. [3]

Following the dissolution of the Soviet Union, interest in LFRs in the Russian Federation has remained strong. This interest is exemplified by some limited work devoted to Accelerator Driven Subcritical (ADS) reactors cooled by LBE and, more importantly, two critical reactor concepts: the LBE-cooled SVBR (Svintsovo Vismutovyi Bystriy Reaktor) and the lead-cooled BREST (Bystriy Reaktor Estestrennoy Bezopasnosti). Both the SVBR and BREST programs are active with near-term (completion dates of 2017 and 2020) construction plans underway.

II.B. European Experience with LBE- and lead-cooled reactors

In Western Europe, initial efforts related to LFR development concentrated on Accelerator Driven Subcritical (ADS) systems for the transmutation of plutonium and minor actinides (MA). Key initiatives included EFIT (European Facility for Industrial Transmutation) and MYRRHA (Multi-purpose hybrid research reactor for high-tech applications). [4] EFIT served as the starting point for the design of later critical reactor systems cooled by lead discussed below while MYRRHA continues as a major project intended to demonstrate both subcritical and critical operation of a system cooled by LBE while operating as a multi-purpose irradiation facility.

In the process of developing these subcritical and critical lead-cooled systems, considerable effort has been spent to exploit to the greatest degree possible the inherent beneficial characteristics of lead or LBE as a coolant while introducing safety design as a primary consideration from the beginning.

The European research program has been developed and funded primarily by the so-called Framework Programs (FWP) of the European Community (EC). Starting with the 5th FWP in 1997, a consortium of major European organizations jointly initiated the development of an ADS prototype as part of the XT-ADS project funded by EC. The subject of transmutation was further investigated in the 6th FWP starting in 2002 through the participation of several major industrial partners and research organizations in a project called IP-EUROTRANS. A major step for the LFR development was taken by the European Lead-cooled System (ELSY) project initiated in 2006. This project aimed to complete a conceptual design of a 600 MWe industrial size plant with challenging objectives in terms of compactness, economy and safety. [5]

In 2010, the previous efforts and the experience gained in these projects were used to initiate two new projects as part of the 7th FWP: the CDT-FASTEF (Central Design Team – fast-spectrum transmutation experimental facility) project and the LEADER (Lead-cooled European Advanced Demonstration Reactor) project.

The CDT-FASTEF project has spent the last three years conducting conceptual design of the MYRRHA facility, an ADS LBE-cooled facility that can be operated in both a subcritical as well as a critical mode, and is envisioned as a pilot plant for LFR technology.

The LEADER project concentrated its activities in the development of an enhanced concept for an industrial-sized critical reactor, the new configuration being designated ELFR (European Lead-cooled Fast Reactor) and, in parallel, undertook design activities for a smaller LFR demonstrator, sized at 120 MWe, designated as ALFRED, the Advanced Lead-cooled Fast Reactor European Demonstrator. Presently, strong efforts of development of ALFRED are underway. [6, 7]

A multiplicity of ancillary R&D projects were also initiated under FP7 to provide support to CDT-FASTEF and LEADER and to address a range of related issues of interest (and synergies) identified in the development of LBE and lead technologies. For example, some of these additional projects are: SARGEN IV (Gen IV safety approach harmonization), SILER (Seismic-Initiated events risk mitigation in Lead-cooled Reactors), MATTER (Materials Testing and Rules), GETMAT (Gen IV and transmutation materials), THINS (Thermal-hydraulics of Innovative Nuclear Systems) and FREYA (Fast Reactor Experiments for hYbrid Applications).

In parallel with these R&D programs, the LFR programme in Europe also benefits from 34 experimental facilities, in operation or under construction, in 10 European research institutions. While such facilities are not enumerated in detail in this paper, it is worthwhile to note that they are dedicated to the main issues identified by LFR designers and are distributed throughout Europe, testimonial to the great interest in numerous countries. As far as GIF is concerned, Europe proposed and promoted the establishment of a Memorandum of Understanding (MOU) related to LFR technology development. This MOU was signed by EC and Japan in 2010, and by Russia in 2011.

II.C. Asian experience with LBE- and lead-cooled reactors

Japan and Korea have also conducted significant research into LFR reactors since at least the 1990s.

In Japan, several interesting projects have been pursued by the Tokyo Institute of Technology. [8] The LSPR (LBE-cooled long-life safe simple small portable proliferation-resistant reactor) system is a small reactor with long-life core, a concept proposed in the early 1990s. This small reactor would be factory fabricated at an energy park, transported to its operating site, and operated for the reactor's life. The reactor would have a sealed vessel which would not be opened at the operating site for refuelling for reasons of proliferation resistance. At the end of the reactor life, it would be removed and replaced by a new one. The old reactor (with its expended fuel) would be shipped back to the nuclear energy park. There would be no residual radioactive waste left at the site. Thus, the operating site and host government or organization would not have to deal with spent fuel or radioactive waste from reactor fuelling operations.

In a separate effort, in 2004 the Tokyo Institute of Technology proposed the PBWFR (Pb-Bi-cooled direct contact Boiling Water Fast Reactor) design concept. [9] This effort evaluated the feasibility of eliminating steam generators and primary pumps by direct injection of feedwater into hot LBE above the core to stimulate coolant circulation. The injected feedwater would boil in the reactor chimney, and steam bubbles would rise with buoyancy force. The resulting bubble motion would serve as the driving force of coolant circulation.

Another interesting concept, the CANDLE (Constant Axial shape of neutron flux, nuclide number densities and power shape) reactor, was also elaborated by the Tokyo Institute of Technology as a sodium-cooled design with a lead-cooled variant [10].

A significant Japanese effort relates to the development of LFR concepts by the Japan Nuclear Cycle Development Institute (JNC, presently JAEA) in their Feasibility Study on the Commercialized Fast Reactor Cycle. [8] In Phase I of this study, typical fast reactor system concepts were identified and compared to different options: coolant types, including lead and lead-bismuth; plant size (i.e., large, medium, and small reactors); tank versus loop designs; and forced versus natural circulation cooling. In Phase II of the study, a concept of a lead-bismuth-cooled, medium size tank type, fast reactor with forced circulation was selected as the preferred LFR, and this concept was investigated to identify its attractive properties as well as drawbacks.

Finally, a notable effort was committed by the Central Research Institute for the Electric Power Industry (CRIEPI) and the Toshiba Corporation to develop the 4S reactor, an innovative small, long-life sodium cooled reactor known as the 4S (Super Safe, Small and Simple) reactor. As part of this effort, some design work was also devoted to the consideration of a lead-cooled variant of this design, sometimes referred to as the L-4S.

Past LFR-related work in Korea has included efforts at the Seoul National University to develop concepts such as PEACER (Proliferation-resistant, Environment-friendly, Accident-tolerant, Continuable, and Economical Reactor) and BORIS. [11, 12]

Of additional note, as described previously, in 2010, Japan joined with EC in signing a MOU on co-operation with respect to LFR technology development.

II.D. U.S. experience with LBE- and lead-cooled reactors

Work on LFR concepts and technology in the U.S. has been carried out from 1997 to the present. During this time frame, work was carried out on lead corrosion and thermal-hydraulic testing at a number of different organizations and laboratories including Los Alamos National Laboratory (LANL), Argonne National Laboratory (ANL) and at the University of Nevada at Las Vegas (UNLV).

At the University of California at Berkeley (UC-B), design work was carried out on the Encapsulated Nuclear Heat Source (ENHS) and related design studies.

Of particular relevance is the development of the design of the Small, Secure Transportable Autonomous Reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organizations over an extended period of time. This concept represents one of the three reference designs of the GIF LFR Provisional System Steering Committee (PSSC) and, as such, is summarized further in a subsequent section of this paper. [13]

It is also worth mentioning that some additional efforts have been carried out or are ongoing in the US. First, alloy and material development studies related to corrosion mitigation and, in particular, the development of the technology of Functionally Graded Composite materials – manufacturable materials that provide protection against corrosion in a molten lead or LBE environment - are being carried out at MIT. [14] In the industrial sector, companies such as GenIV Energy and lakeChime PPRS are also pursuing LFR concepts for commercial application.

III. CURRENT STATUS AND ACTIVITIES OF THE PSSC

The GIF-LFR Provisional System Steering Committee (PSSC) was initially formed in 2005. The original membership included the EC, the US, Japan and Korea. With Korea primarily in observer status between 2005 and 2008, this initial committee worked together to prepare a series of drafts of an initial LFR System Research Plan (LFR-SRP) [15], among its other activities.

During this first phase of the GIF research planning effort, beginning in 2005 and culminating in the completion of the final draft System Research Plan (SRP), two main directions or research thrusts were envisioned: the first was a (relatively) large central station plant for which the reference concept was ELSY [5]; the second was a small transportable LFR system for which the reference concept was SSTAR [13].

In 2010, an MOU was signed between EC and Japan, and this resulted in a reformulation of the PSSC. Then in 2011, the Russian Federation added its signature to the MOU. In April 2012, the reformulated PSSC met in Pisa, Italy and a number of actions were defined. The United States was invited to participate to the activities of PSSC as an observer, and the process of preparing a revised SRP was initiated. The new PSSC, with representatives of EC, Japan and Russia, envisioned various updates to the central station and small reactor thrusts while adding a mid-size LFR (i.e., the BREST-300) as a new thrust in the SRP. In addition, the PSSC decided to prepare a position paper describing the basic advantages and remaining research challenges of the LFR, to be posted on GIF website.

The second meeting of the reformulated PSSC took place on November 7-9, 2012 in Tokyo, hosted by the Tokyo Institute of Technology. This meeting was characterized by a high density of discussions between the members, especially on issues related to material corrosion and on the plant characteristics of the BREST-OD-300, which includes a site-dedicated fuel reprocessing plant. The next meeting of the PSSC is planned to take place in Paris, hosted by OECD-NEA and in conjunction with the IAEA conference on Fast Reactors, FR-13.

The reference concepts that form the basis of the revised SRP activities are the following:

- The European Lead-cooled Fast Reactor (ELFR) for the large, central station plant (600 MWe).
- The BREST-OD-300 (300 MWe) for the medium size plant.
- The Small Secure Transportable Autonomous Reactor (SSTAR – 20 MWe) for the small system.

An overview description of each of these reference systems is provided in the next section.

IV. OVERVIEW DESCRIPTION OF THE THREE REFERENCE LFR SYSTEMS

Beside obvious differences related to the size, the three systems taken as references for the GIF-LFR-PSSC activities share a significant number of technical issues and many common features, especially as far as safety design is concerned; thus, there are many commonalities from the point of view of design and engineering of these systems and the corresponding solutions adopted.

IV.A. The European lead-cooled fast reactor, ELFR

The ELFR system is an evolutionary design representing a modification to the earlier ELSY reactor concept. Figure 1 provides an overview sketch of the ELFR reactor vessel and its contents.

Several of the relevant characteristics of the ELFR design are summarized in Table 1.

Figure 1: ELFR – the European lead-cooled fast reactor

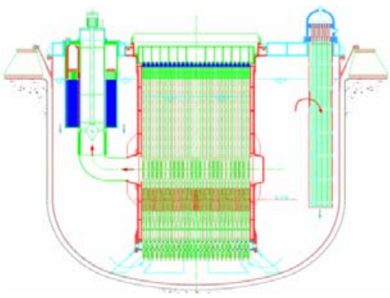


Table 1: ELFR summary parameters

Power	1 500 MW(th), 600 MW(e)
Core diameter	4.5 m
Core height	1.4 m
Core fuel	MOX (1st load)
Coolant temp.	400/480°C
Maximum cladding temp.	550°C
Efficiency	~42%
Core breeding ratio (CBR)	~1

The ELFR primary system has a pool-type configuration, with the main and safety vessels supported by a Y-support holding the main vessel in the upper part. The Reactor Vessel (RV) has been kept as compact as possible, in order to reduce the total coolant inventory and the corresponding seismic loads, while being of sufficient size to accommodate the required number of components [i.e., eight Steam Generators (SGs), eight Primary Pumps (PPs), and eight Decay Heat Dip Coolers (DCs)].

The hot pool of the ELFR vessel is enclosed by an Inner Vessel (IV), connected to the PPs through suction pipes. Each PP is installed at centre of its corresponding SG, which transfers the heat from primary lead coolant to the water-steam in a superheated cycle. The free level of the hot pools inside each SG/PP unit is higher than the free level inside the Inner Vessel, the different heads depending on the pressure losses across component parts of the primary circuit. The design is based on a core pressure loss of 0.9 bar and a total primary pressure loss of 1.4 bar.

The core inlet and outlet temperatures are 400°C and 480°C, respectively, allowing for a sufficient margin in the cold plenum from the freezing point of the lead coolant, while reducing the potential for embrittlement (for structures wetted by cold lead) and corrosion (for structures in hot molten lead).

The maximum speed of the primary coolant is specified at 2 m/s (10 m/s at the tips of the pump impeller blades) in order to limit erosion.

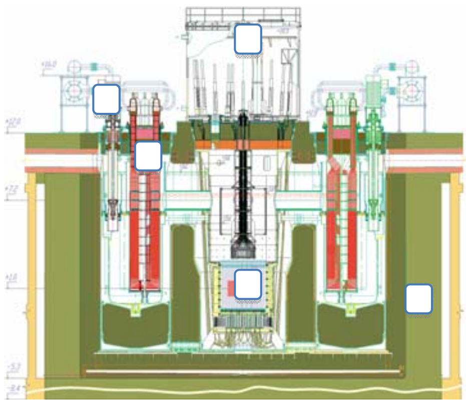
The internal reactor component arrangement and design presents a simple flow path for the primary coolant. The locations of the heat source (within the core) and of the heat sinks (SGs) allow for efficient natural circulation of the coolant under emergency shutdown conditions.

Two safety systems for decay heat removal have been considered as an integral part of the design from the beginning of the activities. They are characterized by passive operation, diversity and redundancy while, in addition, being completely independent from one another.

IV.B. The BREST-OD-300 Russian lead-cooled reactor

The BREST-OD-300 reactor is a pilot demonstration reactor (300 MWe) considered as a prototype of future commercial reactors of the BREST family for large-scale nuclear plants characterized by the idea of "natural safety". Figure 2 provides an overview sketch of the BREST-OD-300 system.

Figure 2: BREST-OD-300



Several of the relevant characteristics of the BREST-OD-300 design are summarized in Table 2.

BREST-OD-300 is a reactor facility of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation circuit with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault.

BREST has a widely-spaced fuel lattice with a large coolant flow area, resulting in low pressure losses, favouring the establishment of primary natural circulation for decay heat removal. It shares with other designs the absence of uranium blankets, replaced by lead reflector with the proper albedo improving power distribution, providing a negative void and density coefficients, and ruling out the production of weapons-grade plutonium. The BREST decay heat removal systems are characterised by passive and time-unlimited residual heat removal directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere.

Table 2: BREST summary parameters

Power	700 MW(th), 300 MW(e)
Core diameter	2.6 m
Core height	1.1 m
Core fuel	UN + PuN
Coolant temp.	420/540°C
Maximum cladding temp.	650°C
Efficiency	43-44%
Core breeding ratio (CBR)	~1

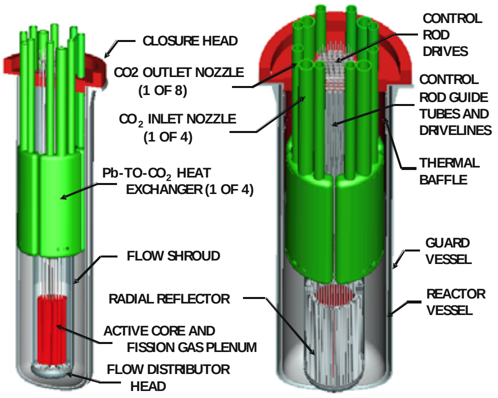
The fuel type considered for the first core of the BREST fast reactor is nitride of depleted uranium mixed with plutonium and Minor Actinides (MA), whose composition corresponds to that of irradiated (spent) fuel from PWR's following reprocessing and subsequent cooling for ~ 20 years.

The characteristics of lead allow for the operation with such fuel at an equilibrium composition. This mode of operation is characterized by full reproduction of fissile nuclides in the core (Core Breeding Ratio (CBR)~1) with irradiated fuel reprocessing in the closed fuel cycle. Reprocessing is limited to the removal of fission products without separating Pu and minor actinides (MA) from the mix (U-Pu-MA). One of the notable characteristics of the BREST plant is that a reprocessing plant is co-located with the reactor, eliminating in principle any accident or problem due to fuel transportation.

IV.C. The small secure transportable autonomous reactor (SSTAR)

SSTAR is a small modular reactor (SMR) that can supply 20 MWe/45 MWt with a reactor system that can be transported in a shipping cask. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life without refueling; and an innovative supercritical CO_2 (S-CO_2) Brayton cycle power conversion system. Figure 3 provides an overview sketch of the SSTAR system.

Figure 3: SSTAR



Several of the relevant characteristics of the SSTAR design are summarized in Table 3.

Table 3: SSTAR summary parameters

Power	**45 MW(t), 20 MW(e)**
Core lifetime	15-30 years
Core fuel	Nitride enriched in N_{15}
Coolant temp.	420/567°C
Maximum cladding temp.	650°C
Efficiency	44%
Core breeding ratio (CBR)	~1

The present pre-conceptual design of SSTAR is that of a small shippable reactor (12 m X 3.2 m vessel), with a 15-30-year life open-lattice cassette core and large-diameter (2.5 cm) fuel pins held by spacer grids welded to control rod guide tubes.

The main mission of the 20MWe (45MWt) SSTAR is to provide incremental energy generation to match the needs of developing nations and remote communities without electrical grid connections, such as those that exist in Alaska or Hawaii, island nations of the Pacific Basin, and elsewhere.

Design features of the reference SSTAR in addition to the lead coolant, 15-30-year cassette core and natural circulation cooling, include autonomous load following without control rod motion, and use of a supercritical CO_2 (S-CO_2) Brayton cycle energy conversion system. The incorporation of inherent thermo-structural feedbacks imparts a high degree of passive safety, while the long-life cartridge core life imparts strong proliferation resistance.

V. RESEARCH CHALLENGES REMAIN

Although many physical characteristics of lead used as a coolant constitute a set of clear advantages with respect to other potential reactor coolants, there are obviously some aspects that need specific developments. LFR research challenges are mainly related to the following aspects: the high melting point of lead; its opacity; coolant mass; and potential for corrosion of structural steels.

The high melting temperature of lead (327°C) requires that the primary coolant system be maintained at sufficiently high temperatures to prevent solidification. This presents design as well as engineering challenges during operation and maintenance, although is not considered by designers a safety issue.

The opacity of lead, in combination with its high melting temperature, presents challenges related to inspection and monitoring of reactor in-core components as well as fuel handling. Important synergies are however possible with SFR technology, where specific developments are underway. In addition, design innovation can reduce needs in this area; for example, with the ELFR system, fuel element extension into the cover gas above the free surface of the coolant offers the possibility to directly monitor fuel status under more favorable conditions.

The high density and corresponding high mass of lead require careful consideration of structural and seismic design. This issue can be addressed by the adoption of technology such as seismic isolation as is being done in design-specific projects related to ELFR.

Significant challenges result from the phenomena related to lead corrosion of structural steels at high temperatures and flow rates. These phenomena require careful material selection and component and system monitoring during plant operations. In the past, Russian scientists developed the technology of continuous passivation of the structural steels based on their early LFR experience and were able to solve the problems encountered during early reactor deployments. Many efforts of a fundamental nature are being carried out in several European laboratories in order to investigate specific aspects and peculiarities of this technology.

VI. LFR AND EXTREME NATURAL EVENTS

In the development of any reactor concept, safety is a critical consideration. Following the events related to the natural disaster and reactor accidents at Fukushima-Daiichi, it is especially important to consider the ability of reactor systems to respond successfully to extreme events.

The use of lead (or LBE) as a coolant offers several important advantages in this regard. In an interesting recent analysis, Toshinsky et al. [16] considered the question of stored energy in various types of reactor coolants. This analysis highlights several important advantages of lead coolants with respect to the availability of energy (thermal, pressure-related or chemical) to exacerbate accident conditions whether initiated by natural phenomena or not. As low-pressure coolants with relative chemical inertness, lead and LBE have innate characteristics that enable a high level of passive safety and suggest more benign end states in the case of unforeseen accident conditions. In the following paragraphs, some particular aspects of LFR safety approaches and characteristics are identified.

Seismic and structural designs are important considerations for any reactor system, and especially larger systems. In the case of the current LFR system design for the largest reference system, ELFR, we note that the ability to respond to earthquakes has been significantly enhanced by the adoption of seismic isolation.

With respect to Decay Heat Removal (DHR) Systems, the current reference designs feature independent, redundant, diverse, and completely passive DHR systems. Only actuation (through valve alignment) is active, and this would use local stored energy. As a result, station blackout – which was a critical factor at the Fukushima-Daiichi accidents – would not present a threat; any initiating event would be managed without requiring AC power or other external energy source.

Even if one postulates station blackout without functional DHR systems available, recent safety analyses for ELFR demonstrated that fuel and cladding temperatures would not reach critical levels. In fact, safety analyses are now normally performed assuming the absence of off-site power.

A complete core melt would be extremely unlikely due to favorable intrinsic lead characteristics: its high thermal inertia and very high boiling point would limit the possibility for fuel melting, and the higher density of the coolant in contrast with the oxide fuel would result in fuel dispersion in contrast to fuel compaction if an otherwise unforeseen event resulted in fuel disruption.

Furthermore, in the very unlikely event of an extreme Fukushima-like scenario (or beyond) leading to the loss of all heat sinks (i.e., loss of both DHR and secondary systems), reactor heat could still be removed by water cooling of the cavity between the reactor primary and safety vessels, while in the extreme case of a reactor primary vessel failure, additional systems external to the safety vessel can be envisioned to cool the lead pool. This scenario would imply a highly extreme situation in which all heat sinks of the system had been already lost. A main advantage of the LFR is that in such an extreme, beyond design condition a highly abundant and readily available fluid (i.e., water) can be used to cool the reactor and retain the system in a safe condition. The benign behaviour of the LFR in response to such extreme (and unanticipated) events is enabled by the inherent, natural characteristics of the coolant and is not dependent on complex or engineered features that could be subject to failure.

Thus, the reactor systems envisioned as references for the GIF-LFR-PSSC research activities present highly promising behavior when challenged by extreme events. The physical and chemical characteristics of lead as a coolant provide very advantageous safety feedbacks that have been leveraged by designers to further enhance system response to any envisaged transient.

VII. CONCLUSIONS

In conclusion, lead- and LBE-cooled systems offer great promise in terms of fast reactor plant simplification, performance and safety response while offering sustainability advantages common to other fast reactor systems.

The Russian experience with the deployment of LBE-cooled systems for submarine propulsion provided an excellent demonstration that the LFR can be produced and operated on an industrial scale.

Still, additional work is needed to achieve commercial deployment of new commercial LFR systems. Some important areas for R&D include:

- Completion of designs of commercializable systems as well as demonstration systems.
- Testing of special materials for use in lead environment.
- Completion of fuel studies, including recycle.
- Special studies (e.g., studies related to seismic response; sloshing; LBE dust/slag formation).
- Evaluation of long-term radioactive residues from fuel and system activation.
- Technology pilot plant/Demo activities.

Finally, in the post-Fukushima environment, the unique safety potential of the LFR should be recognized and leveraged.

NOMENCLATURE

ADS	Accelerator Driven Subcritical
ALFRED	Advanced Lead-cooled Fast Reactor European Demonstrator
ANL	Argonne National Laboratory
BREST	Bystriy Reaktor Estestrennoy Bezopasnosti
CANDLE	Constant Axial Neutron flux, densities and power shape During Life of Energy
CBR	Core Breeding Ratio
CDT-FASTEF	Central Design Team for a fast-spectrum transmutation experimental facility
CRIEPI	Central Research Institute for the Electric Power Industry
DC	Decay Heat Dip Cooler
DHR	Decay Heat Removal
EC	European Community
EFIT	European Facility for Industrial Transmutation
ELFR	European Lead-cooled Fast Reactor
ELSY	European Lead-cooled System
FA	Fuel Assembly
FREYA	Fast Reactor Experiments for hYbrid Applications
FWP	Framework Program
GETMAT	Gen IV and transmutation materials
GIF	Generation IV International Forum
IV	Inner Vessel
JAERI	Japan Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LBE	Lead-Bismuth Eutectic
LEADER	Lead-cooled European Advanced Demonstration Reactor
LFR	Lead-Cooled Fast Reactor
LLNL	Lawrence Livermore National Laboratory
LSPR	LBE-cooled long-life Safe Simple Small Portable Proliferation resistant Reactor
ISI	In-Service Inspection
MA	Minor Actinide
MATTER	Materials Testing and Rules
MIT	Massachusetts Institute of Technology
MOU	Memorandum of Understanding
MYRRHA	Multi-purpose hybrid research reactor for high-tech applications
NERAC	U.S. Nuclear Energy Research Advisory Committee
PBWFR	Pb-Bi-cooled direct contact boiling Water Fast Reactor
PEACER	Proliferation-resistant, Environment-friendly, Accident-tolerant, Continuable, and Economical Reactor
PP	Primary Pump
PSSC	Provisional System Steering Committee
RV	Reactor Vessel

RVACS	Reactor Vessel Air Cooling System
SARGEN IV	Gen IV safety approach harmonization
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SILER	Seismic-Initiated events risk mitigation in Lead-cooled Reactors
SMR	Small Modular Reactor
SRP	System Research Plan
SSTAR	Small Secure Transportable Autonomous Reactor
SVBR	Svintsovo Vismutovyi Bystriy Reaktor
THINS	Thermal-hydraulics of Innovative Nuclear Systems
UNLV	University of Nevada at Las Vegas

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THE MOLTEN SALT REACTOR (MSR) IN GENERATION IV: OVERVIEW AND PERSPECTIVES

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ABSTRACT

The MSR is distinguished by its core in which the fuel is dissolved in molten fluoride salt. The technology was first studied more than 50 years ago. Modern interest is on fast reactor concepts as a long term alternative to solid-fuelled fast neutrons reactors. The onsite fuel reprocessing unit using pyrochemistry allows breeding plutonium or uranium-233 from thorium. R&D progresses toward resolving feasibility issues and assessing safety and performance of the design concepts. Key feasibility issues focus on a dedicated safety approach and the development of salt redox potential measurement and control tools in order to limit corrosion rate of structural materials. Further work on the batch-wise online salt processing is required. Much work is needed on molten salt technology and related equipments.

I. INTRODUCTION

Molten Salt Reactor (MSR) technology was partly developed, including two demonstration reactors, in the 1950s and 1960s in the USA (Oak Ridge National Laboratory). The demonstration MSRs were 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) with those relating to molten salt fluorides as fluid fuel and coolant (low pressure and high boiling temperature, optical transparency) [1-5].

In contrast to most other molten salt reactors previously studied, the MSFR does not include any solid moderator (usually graphite) in the core. This design choice is motivated by the study of parameters such as feedback coefficient, breeding ratio, graphite lifespan and ^{233}U initial inventory. MSFR exhibit large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors.

Compared with solid-fuel reactors, MSFR systems have lower fissile inventories, no radiation damage constraint on attainable fuel burn-up, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics give MSFRs potentially unique capabilities for actinide burning and extending fuel resources.

MSR developments in Russia [6, 7] on the Molten Salt Actinide Recycler and Transmuter (MOSART) aim to be used as efficient burners of transuranic (TRU) waste from spent UOX and MOX light water reactor (LWR) fuel without any uranium and thorium support and also with it. Other advanced reactor concepts are being studied [8, 9], which use the liquid salt technology, as a primary coolant for Fluoride salt-cooled High-temperature Reactors (FHRs), and coated particle fuels similar to high temperature gas-cooled reactors.

More generally, there has been a significant renewal of interest in the use of liquid salt as a coolant for nuclear and non-nuclear applications [2, 10]. These salts could facilitate heat transfer for nuclear hydrogen production concepts, concentrated solar electricity generation, oil refineries, and shale oil processing facilities amongst other applications.

The paper provides an overview of the main technical activities 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

The decision for setting up a Provisional System Steering Committee (PSSC) for the MSR with Euratom, France, the Russian Federation and United States was taken by the GIF Policy Group in May 2004. In 2009 discussions were held on the mode of co-operation on MSR R&D in GIF. The Policy Group took the decision to set up a Memorandum of Understanding (MOU) for both the MSR and LFR systems. This MOU would provide a more flexible structure for R&D co-operation on those systems in the GIF framework for the mid-term. The MOU has been signed by France and JRC, on behalf of Euratom, October the 6th 2010. The United States and Russia will remain observers, but Russia is considering signing the MOU in the medium term future.

The members of the PSSC MSR, France and Europe, are working on MSFR (Molten Salt Fast Reactor) in which the salt is the fuel and the coolant. The common objective of these projects is to develop a conceptual design for an MSFR with an effective system configuration – resulting from physical, chemical and material studies – for the reactor core, the reprocessing unit and waste conditioning. The conceptual design activities are intended to increase the confidence that MSFR systems can satisfy the goals of Generation IV reactors in terms of sustainability (Th breeder), non proliferation (integrated fuel cycle, multi-recycling of actinides), resource savings (closed Th/U fuel cycle, no uranium enrichment), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner).

Russia, which participates in the PSSC as an observer, works on flexible MOSART (Molten Salt Actinide Recycler & Transmuter) system fuelled with different compositions of plutonium and minor actinide (MA) trifluorides with and without Th support. The United States, which participates in the PSSC as an observer mainly works on FHRs (Fluoride-salt-cooled high temperature reactor) as a nearer term reactor class whose technology developments are supportive of MSFRs.

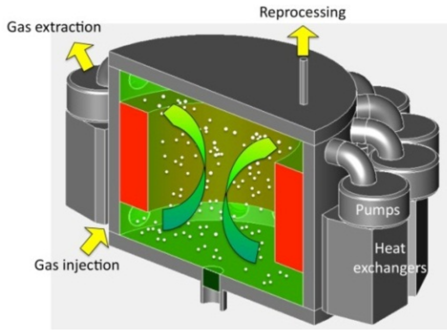
III. MSR CONCEPTS

Two reactors concepts using molten salts are studied in the GIF molten salt reactor provisional system steering committee, i) molten salt reactors, in which the salt serves as both the fuel and the coolant, and ii) reactors with solid fuel cooled by molten fluoride salts.

- MSFR concept

Recent conceptual developments on fast neutron spectrum molten salt reactors (MSFRs) using fluoride salts open promising possibilities to exploit the ^{232}Th-^{233}U cycle. On the other hand, they can also contribute to significantly diminishing the radiotoxic inventory from present-reactor spent fuels, in particular, by lowering the mass of transuranic elements (TRU).

Figure 1: Schematic conceptual MSFR design



In the MSFR, the liquid fuel processing is performed on a small side stream of the molten salt. Fission products are removed from the side stream and the remainder is then returned to the reactor. This is fundamentally different from a solid fuel reactor where separate facilities produce the solid fuel and process the used nuclear fuel. Because of this design characteristic compared to classical solid-fuel reactors, the MSFR can thus operate with widely varying fuel compositions.

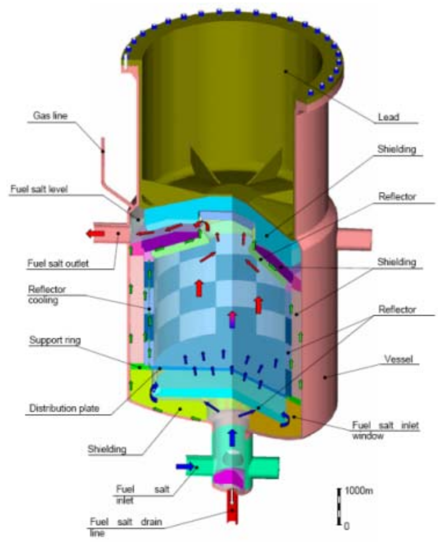
Figure 1 sketches schematically general outlines for such a MSFR. The core consists of moving fuel loaded fluoride salt (note the lack of graphite moderation in core). The reference MSFR is a 3 000 MWth reactor with a total fuel salt volume of 18 m^3, operated at a mean fuel temperature of 750°C. The salt is composed of lithium fluoride and thorium fluoride and the proportion of heavy nuclei is fixed at 22.5%. In preliminary drawings done in relation to calculations, the core of the MSFR is a single compact cylinder (2.25m high x 2.25m diameter) where the nuclear reactions occur within the liquid fluoride fuel salt acting also as the coolant.

The fuel salt flows freely from the bottom to the top of the central part of the core without any solid moderator. The return path of the salt (from the top to bottom) is divided into 16 sets of pumps and heat exchangers located around the core. Bubbles are injected in the fuel salt circulation after the exchangers and separated from the liquid at the core outputs. The fuel salt runs through the total cycle in 3-4 seconds. The total fuel salt volume is distributed half in the core and half in the external fuel circuit (salt collectors, salt-bubble separators, fuel heat exchangers, pumps, salt injectors and pipes). The lower neutronic reflector is connected to a drain system enabling the reactor core to be drained for planned shut downs or in case of incident/accident that leads to a temperature increase in the core. Thus the entire fuel inventory can be passively drained by gravity into subcritical, passively cooled tanks.

- Molten Salt Actinide Recycler and Transmuter (MOSART) concept

MSR developments in Russia on the 2 400MWt Molten-Salt Actinide Recycler and Transmuter (MOSART) address the concept of large power units with a fast neutron spectrum in the core [6]. Promising configuration for 2 400 MWt MOSART is the homogeneous cylindrical core (3.6 m high and 3.4 m in diameter) with 0.2 m graphite reflector filled with 100% of $73LiF\text{-}27BeF_2$ salt mixture. It is feasible to design critical homogeneous core fuelled only by TRU trifluorides from UOX or MOX LWR used fuel while equilibrium concentration for trifluorides of actinides (0.4 mole% for Li,Be/F core, with the rare earth removal cycle 300 efpd) is truly below solubility limit (~2 mole%) at minimal fuel salt temperature in primary circuit 600-620ºC. Recently [22], the flexibility of single fluid MOSART concept fuel cycle is underlined, particularly, possibility of its operation in self-sustainable mode using different loadings and make up. Single fluid 2 400MWt Li, Be/F MOSART core containing in initial loading 2 mole% of ThF_4 and 1.2 mole% of $TRUF_3$, with the LnF_3 removal cycle 300 efpd after 12 years of slow increasing of Th content in the solvent can operate without any $TRUF_3$ make up basing only on Th support as a self-sustainable system. The maximum concentration of TRU during this transition does not exceed 1.7 mole%. At equilibrium molar fraction of fertile material in the fuel salt is near 6% and it is enough to support the system with CR=1 within 50 years reactor lifetime.

Figure 2: Molten salt actinide recycler and transmuter (MOSART) concept

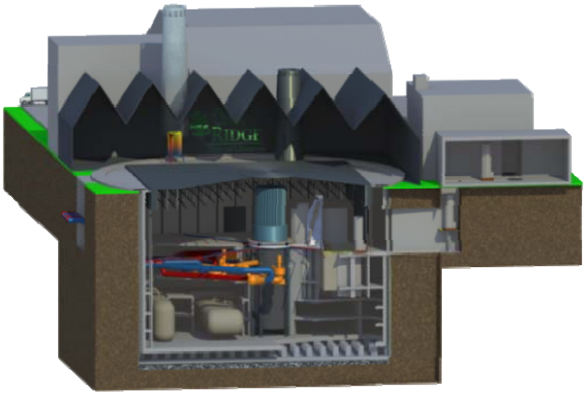


- Fluoride-salt-cooled high-temperature reactor concept

Several different FHRs are currently under design by different organizations. Oak Ridge National Laboratory (ORNL) is leading the preconceptual design of the Advanced High Temperature Reactor (AHTR – see Figure 3), which is a large [1 500 MW(e)] central station-type power plant focused on low-cost electricity production. The Massachusetts Institute of Technology (MIT) is leading the efforts toward developing a preconceptual design for a <20 MW(t) test reactor. The Shanghai Institute of Technology is leading a design effort to develop the first FHR critical facility/test reactor [2 MW(t)]. The University of California at Berkeley is developing a preconceptual design for a mid-sized [410 MW(e)] initial commercial prototype reactor. ORNL is also developing a preconceptual design of a Small modular Advanced High-Temperature Reactor [SmAHTR;

125 MW(t)] focused on thermal power production.

Figure 3: AHTR reactor building layout overview



FHRs, in principle, have the potential to be low-cost energy producers while maintaining full passive safety. FHRs do not require any system or operator active response to avoid core damage or large off-site release of radioactive material for any design basis accident or non low-frequency beyond design basis accident, including severe earthquakes, tsunamis, large commercial plane impact, or permanent station blackout. The safety characteristics of FHRs arise from fundamental physics as well as well-designed, constructed, and maintained systems, structures, and components (SSCs). As with other high-temperature plants, FHRs can efficiently produce both electricity and process heat, including effective support for liquid hydrocarbon fuel production.

FHRs are a research focus of the U.S. Department of Energy's advanced reactor concepts program. The U.S. program includes technology development and demonstration as well as concept design studies focused on a test reactor, small modular reactors, and large-scale power plants. FHRs are also included within the research plans of both the Chinese and Indian civilian nuclear energy programs.

IV. R&D OBJECTIVES AND PROGRESS WITHIN PSSC-MSR

The common objective of these projects is to propose a conceptual design of MSFR as the best system configuration – resulting from physical, chemical and material studies – for the reactor core, the reprocessing unit and wastes conditioning. It is intended to deepen the demonstration that the MSFR system can satisfy the goals of Generation IV. Those topics are the subject of the following sub-sections:

- Physical studies
- Safety
- Materials studies
- Salt properties
- Salt reprocessing
- Technological studies

IV.A. Physical studies

Feedback coefficient evaluation

The potential of MSFRs without moderator in the core leading to a fast neutron spectrum while ensuring excellent safety coefficients was highlighted [1]. Various MSR configurations were studies by modulating the amount of graphite in core to obtain a thermal, an epithermal, or a fast spectrum. In particular, configurations of a fast spectrum MSR (MSFR) have been identified with outstanding safety characteristics and minimal fuel-reprocessing requirements. It has very negative feedback coefficients. This is true not only for the global temperature coefficient but also for the partial coefficients that characterize the dilatation or the heating of the salt, and the void effect.

First core and deployment capacities

Studies of the different starting modes of the MSFR have been performed [12, 13]. The MSFR concept may use as initial fissile load, ^{233}U or uranium or also the transuranic elements currently produced by light water reactors. The characteristics of these different launching modes of the MSFR and the thorium fuel cycle have been studied, in terms of safety, proliferation, breeding, and deployment capacities of these reactor configurations.

Studies show that the MSFR configurations corresponding to various starting modes of the reactor are all characterized by excellent safety coefficients and have the same good deployment capacities. Optimizing the specific power in the MSFR configuration started directly with ^{233}U as initial fissile matter has allowed a reduction of the initial fissile inventory down to three metric tons per GWe. The MSFR is characterized by a low proportion of minor actinides in the salt (around one percent at equilibrium) and by its excellent safety coefficients (-5 pcm/°C).

^{233}U does not exist on earth and is not being directly produced today. The possibility of using in MSFR the transuranic elements (TRU) currently produced in the world as an initial fissile load has been investigated. MSFRs can be started with the Pu+Minor Actinides (TRU) extracted from used UOX fuel discharged from LWR reactors. The TRU-started MSFR is able to efficiently convert the plutonium and minor actinides from generation II-III reactors in ^{233}U while improving the deployment capabilities of the MSFR concept. A transition can be effected to the $^{232}Th/^{233}U$ cycle. The time scale for an almost complete transition is approximately one century. Its only drawback lies in its high initial plutonium concentration above its estimated solubility limit in the $LiF\text{-}ThF_4$ reference salt. To overcome this limitation while still using TRU elements in the initial fissile load of the MSFR to close the current fuel cycle, two optimized solutions have been proposed: mixing the TRU elements at a lower concentration (around 3 to 4 mole%) with either natural uranium with an enrichment ratio of 13% or ^{233}U produced in other reactors.

Coupling of neutronic and reprocessing simulation codes

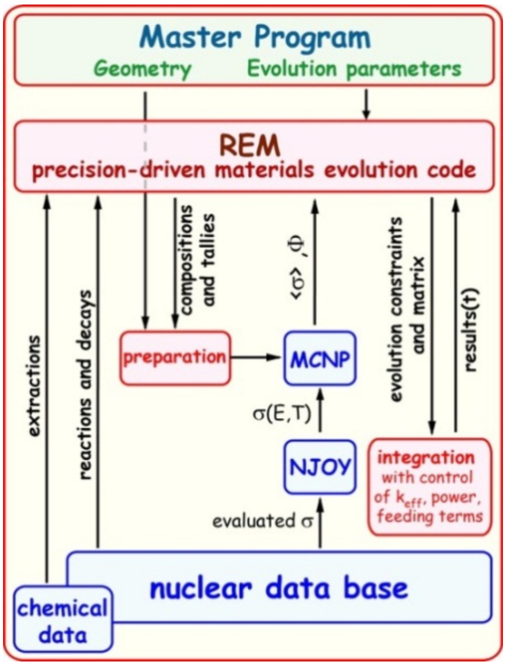
Essentially, because the salt is the moderator, the coolant and the fuel, the study of MSFR are specific. There are strong coupling between neutronics and other part of physic field like the chemistry for instance. Simulations of the MSFR concept rely on numerical tools making use of the MCNP neutron transport code coupled with a homemade materials evolution code [14, 15] (see Figure 4).

The coupling of neutronic and reprocessing simulation codes in a numerical tool has been used to calculate the extraction efficiencies of fission products, their location in the whole system (reactor and reprocessing unit) and radioprotection issues.

Preliminary results based on rough data of the pyrochemical processes involved, illustrate the potential of the neutronic-reprocessing coupling that has been developed. Studies are however still limited by the uncertainties on the design and knowledge of the chemical reprocessing processes.

Detailed neutronic and thermal-hydraulic behaviors of the core have been considered up to now with simplified coupling approach. Actions are under progress at the present time through EVOL projet at the European scale and through a local collaboration in France to develop a 3D coupled model for the MSFR core.

Figure 4: Coupling scheme of the MCNP neutron transport code with the in-house materials evolution code



IV.B. Safety

Molten salt reactors are liquid fuel reactors so that they are flexible in operation but very different in the safety approach from solid fuel reactors. Since this new nuclear technology is in development, safety is an essential point to be considered all along the R&D studies. The first step of the safety approach is a systematic description of the MSFR, limited to the main systems surrounding the core [16].

Thanks to the negative reactivity feedback coefficient, the main scenarios lead to a reactor shutdown.

In order to assess the behavior of the fuel salt after reactor shutdown, a tool to calculate the decay heat has been developed and validated. It can be concluded that the decay heat in the core and the fuel loops of the MSFR is relatively low (3.5% of nominal power compared to 6% in a PWR) primarily thanks to the reprocessing system. The fission products that remain in the core contribute to the fuel salt heating up to 3% of nominal power. An important part of the decay heat (around 2% of nominal power) is located in the reprocessing units, mainly in the gas reprocessing unit, so that its safety assessment should be studied separately. The actinides also have an important contribution (0.5% of nominal power), that becomes dominant some hours after reactor shut down.

With a tool based on point kinetics, loss of heat sink transients can be calculated and their impact on the fuel salt temperature studied. The results of this study demonstrate the importance of the inertia of the systems. We conclude that slow transients (> 1 minute), thanks to a large system inertia, are advantageous and that, with them, the fuel salt temperature increase is slower.

These residual heat calculations will be the basis for the design of the draining system, as drainage must occur for any reactor shut down, whether in normal or in accidental conditions. The impact of the stagnant heating fuel salt on the core and fuel loop systems will be studied as well. It appears that slow transients are favorable (> 1min) to minimize the temperature increase of the fuel salt.

IV.C. Material studies

The structural materials retained for MSR container are Ni-based alloys with a low concentration of Cr. The composition of the alloy was optimized by ORNL researchers for corrosion resistance (both in a low oxygen gas atmosphere and in molten fluorides), irradiation resistance and high temperature mechanical properties. The composition of this optimized Hastelloy N (Ni- 8wt% Cr-12wt% Mo) proved satisfactory up to 750°C, a temperature in the low range of the MFSR. The operating temperatures chosen in neutronic calculations of MSFR systems are ranged between 700 and 850°C. Due to the evolving microstructure of optimized Hastelloy N at higher temperature, it would be impossible to preserve the required material properties in the full operating temperature range required for the MSFR system.

For this high temperature domain, the replacement of Mo by W could prove beneficial for mechanical properties since tungsten diffusion is roughly ten times slower in nickel than molybdenum diffusion. A better creep resistance is expected with a Ni-W solid solution than with a Ni-Mo solid solution. This would help to reach higher in-service temperature. First results show that such material have the required properties, especially in terms of compatibility with molten salts and mechanical properties.

Experimental studies focus on the potential for using Ni-W-Cr alloys as structural materials for MSFR system. The corrosion of a specific Ni-25W-6Cr (wt.%) alloy was studied in a LiF-NaF molten salt, at 750°C and 900°C, for 350 h and 900 h. The results showed, as expected, a selective oxidation of Cr in the alloy. They also evidenced a noticeable and unexpected corrosion of W, which might be attributed to the combined presence of some pollution (by O^{2-} and Fe^{2+} ions) in the salt.

It has been demonstrated that the salt redox potential is a key parameter in the corrosion phenomena of structural materials of MSRs [10, 17]. 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 U(IV)/U(III). This 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. An innovative method (scanning electrochemical micro-scopy SECM) has been proposed to improve the understanding of the corrosion mechanisms at a microscopic scale. Efforts are also made on reactor vessel design to suppress the highest temperature points, and protect or cool some areas. This improvement process will be correlated to a correct 3D simulation of the core.

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. Its metallurgy and in-service properties need to be investigated in further details regarding irradiation resistance and industrialization. Additional tests are being carried our in order to *better understand the W behavior and eventually suppress its corrosion using a highly purified solvent. A special attention will have to be paid to the measure and control of the U(IV)/U(III) ratio in order to reach the desired corrosion resistance of Ni based alloys.*

IV.D. Salt properties

Thermodynamic properties of the salt systems are investigated (JRC/ITU) in order to collect new data which are necessary for developments of molten salt reactor designs, reprocessing scheme and simulation codes [18, 19]. A strong tool can be found in the assessment of phase diagrams. This method is based on the Gibbs free energy minimization between the different phases. With a good description of the phase diagram it is also possible to predict some properties, e.g. vapor pressure, for which no experimental information is available. Determination and modeling of molten fluoride properties salts are developed in parallel with experimental facilities:

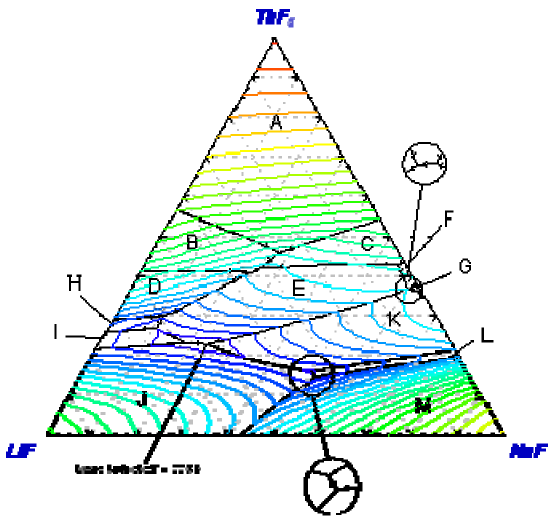
- Alpha glove box which for synthesis and purification of actinide fluorides.

- A Raman spectrometer set up for measurements of the Raman spectra of molten salts. This set-up will allow the determination of the local structure of the actinides in the fluoride salts.

Experimental investigation of physico-chemical properties of actinide fluorides containing salts are carried out. To elucidate the influence of the salt composition on thermo-physical properties of the MSR fuel (melting temperature, solubility of actinides and vapour pressure), it is necessary to understand the phase equilibria in the fuel system. The thermodynamic properties of all phases considered in a multi-component system such as the MSR fuel have to be assessed. Extensive thermodynamic database of various fluoride systems is thus being developed.

Among these properties the heat capacity, is especially important for the heat transfer evaluations within the various loops of MSR. Using a drop calorimetry, a systematic study of the heat capacity of binary LiF-AlkF (Alk = Na, K, Rb, Cs) systems has been finalized. Based on these results it appears that increased heat capacity can be expected in multi-component fluoride mixtures compared to its pure components contributing to higher safety of MSR as the higher the heat capacity the higher the buffer zone for overheating of a reactor during off-normal or accidental conditions.

Figure 5: The calculated LiF-NaF-ThF_4 pseudo-ternary phase diagram with fixed concentration of UF_4 set to 2.55 mol %



Novel technique to measure mixing enthalpies of fluoride liquid solutions using a differential scanning calorimeter has been developed and first tested on the LiF-KF system showing excellent agreement to literature values. Using this promising technique mixing enthalpies of the LiF-ThF_4 system was first measured and the fusion enthalpy of the Li_3ThF_7 intermediate compound was determined. Furthermore, from this experimental campaign new phase diagram data points of the whole LiF-ThF_4 system was obtained.

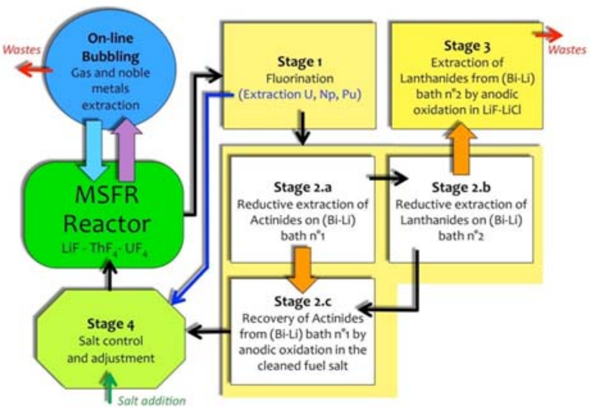
The effort dedicated to the construction of the thermodynamic database that is being developed at ITU since 2002 will continue with the acquisition of data on TRU elements.

IV.E. Salt reprocessing

The on-site salt management of the MSFR combines a salt control unit, an on-line gaseous extraction system (see IV.F Molten salt technological studies) and an offline lanthanide extraction component by pyrochemistry. This salt reprocessing scheme is presented in Figure 6.

The salt properties and composition are monitored through the on-line chemistry control and adjustment unit. A fraction of salt is periodically withdrawn and reprocessed off-line in order to extract the lanthanides before it is sent back to the core. In this separate batch reprocessing unit 99% of uranium (including ^{233}U) and neptunium, 90% of plutonium are extracted by fluorination and immediately reintroduced in the core. The remaining actinides are then quickly extracted together with Protactinium and also sent back to the core. Finally, the lanthanides are separated from the salt through a second reductive extraction and sent to waste disposal.

Figure 6: MSFR reprocessing scheme



The reference scheme depicted in Figure 6 involves four stages for the batch on-site fuel processing [20, 21]. The peculiarity of the concept appears in stages two and three 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. For the core of the flow sheet, the process proposed is a reductive extraction using a liquid metal solvent. Some analytical relations have been established (considering experimental redox potentials and activity coefficients in molten salt and liquid metal) to understand the influence of the liquid solvent composition on the extraction efficiency. The liquid metal is constituted of Bi which is the metallic solvent and of Li which is the reductive reactant.

The experimental tests of extraction process require an optimized procedure for the preparation of the metallic phase. The composition of the metallic phase is a key point for the extraction efficiency. Different procedures of metallic phases preparations have been tested. The method retained for the preparation of the metallic phase is the electrolysis of LiCl-KCl.

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. This value is almost two orders of magnitude less than the reference MSBR scheme.

Acquisition of fundamental data for the extraction processes is still 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.

IV.F. Molten salt technological studies

The gaseous extraction system is a continuous salt chemistry process. Helium bubbles are injected at the lower part of the core to trap the non-soluble fission products (noble metals) dispersed in the flowing liquid as well as the gaseous fission products. A liquid/gas phase separation is then performed on the salt flowing out of the core to extract gaseous species and dragged condensed particles. Following this "physical" process of purification, a small part of the gas is withdrawn to let the fission products decay, and the remaining part of gas is sent back to the lower part of the core.

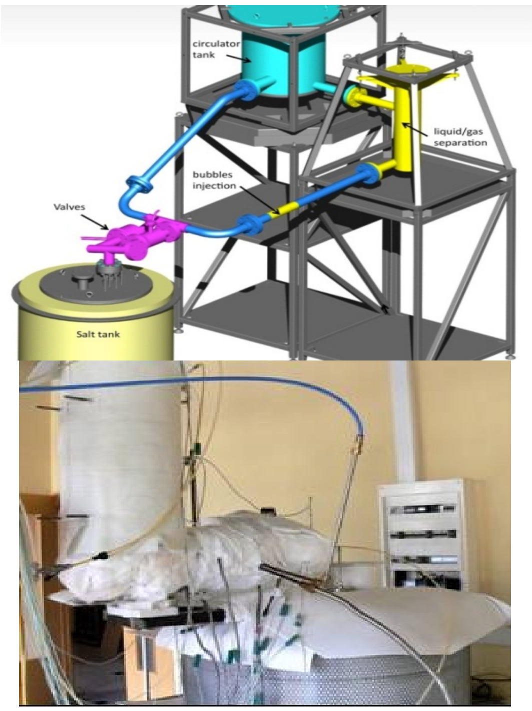
Bubbling treatment needs the insertion of an injector and a liquid gas separator in the salt circuit between the core and heat exchangers. In order to begin the conception of the bubbling components for reactor scale, an experimental project was launched, based on the construction of a molten salt loop (Forced Fluoride Flow for Experimental Research project – LPSC, Grenoble, France). FFFER is dedicated to bubbling studies and is operated with LiF-NaF-KF salt.

Studies dedicated to bubbling cleanup process have led to the conception of a liquid-gas separator with satisfying efficiency when measured on water mock up. The volumic gas rate domain investigated is between 0.02 to 0.5%. Specific features determined on the mock up are reported on the metallic separator internal design. The whole design of the loop has been then achieved. It is presented in Figure 7 together with a picture of the FFFR partial running test. The loop tank separation system is composed of two parts in parallel connection, a metallic valve and a cold plug. The salt tank can contain up to 100 liters but the loop circuit is designed for running with a volume ranging from 50 to 80 liters.

Fabrication of the salt mixture (LiF-NaF-KF) to be used in the French molten salt loop (FFFER project) has been achieved. Tests with liquid salt have been undertaken to prove the ability of our cold plug system to play the role of a security valve on the loop circuit. Satisfying results have been obtained; modifications on the first cold plug design have been done to improve the resistance to corrosion of the whole component. Further evolutions of this component will be made on a separated system in glove boxe, to explore other design possibilities and to acquire data for simulation.

Implementation of instrumentation (temperature, level and flow rate measurement) on the whole experimental set up is under progress. The start of the loop running is foreseen for the middle of year 2013. The future R&D studies will focus on gas and particles extraction efficiency (gas/salt separation, gas analysis by mass spectrometry).

Figure 7: Forced fluoride flow for experimental research project (FFFER)



V. R&D PROGRESS FOR MOSART AND FHR CONCEPTS

MOSART

In Russia, study is under progress within ISTC#3749 and MARS projects to examine the conceptual feasibility of flexible Molten Salt Actinide Recycler & Transmuter (MOSART) system fuelled with different compositions of plutonium and Minor Actinide (MA) trifluorides with and without Th support [22]. New fast-spectrum design options and salt compositions with adequate solubility for actinide trifluorides are being examined with objective to obtain reliable and abundant source of energy through efficient use of transuranium elements from used LWR fuel as well as uranium and thorium resources. Experimental data base created within the projects is used for further development of technology as applied to consumption of actinides while extracting their energy.

Key thermal physical and chemical properties of molten binary LiF-BeF_2, LiF-NaF, and LiF-ThF_4, ternary LiF-BeF_2-ThF_4 and LiF-NaF-KF mixtures important for the design calculation were experimentally studied. Melting temperatures, plutonium and americium trifluorides solubility for the mentioned above salt solvent systems are measured. New experimental data on viscosity, density, thermal conductivity and heat capacity for selected molten binary and ternary salts are received in temperature range from liquidus temperatures till to 750°C.

Particularly, for 78LiF-22ThF_4 (mole%) fuel solvent systems used both in MOSART and MSFR designs following experimental dependences on the PuF_3 solubility (lgS, logarithm of PuF_3 molar concentration), density (ρ,g/cm^3), thermal conductivity (λ,W·m^{-1}·K^{-1}) , heat capacity (c_p,J·g^{-1} K^{-1}) and viscosity (ν, 10^{-6} m^2/s) vs. temperature (T,K) in range from liquidus up to 1 100K were, respectively, obtained [23]:

$$\lg S = 2.58 - 1\,733/T$$
$$\rho = 4.742 - 8.82\cdot 10^{-4}\,T$$
$$\lambda = 0.928 + 8.397\cdot 10^{-5} T$$
$$c_p = -1.111 + 0.00278\times T$$
$$\nu = 1.9798 \exp\{3689 \times (1/T - 0.9698E\text{-}3)\}$$

Electrochemical behavior of the dissolved trifluorides in molten LiF-BeF_2, LiF-ThF_4, and LiF-BeF_2-ThF_4 solvent systems selected are studied. New experimental data on reductive extraction of the lanthanum, neodymium and thorium for the molten salt/liquid bismuth systems at 650°C are obtained. The measured distribution coefficients are consistent with the earlier data obtained for binary $LiF\text{-}BeF_2$ and $LiF\text{-}ThF_4$ systems as well as for ternary $LiF\text{-}BeF_2\text{-}ThF_4$ salt mixtures. The distribution coefficients obtained for $LiF\text{-}ThF_4$ and $LiF\text{-}BeF_2\text{-}ThF_4$ salts with relatively high concentration of ThF_4 (> 20 mole%) cannot provide the effective separation between thorium and lanthanides in the fluoride salt/bismuth solutions. Excellent separation of thorium from lanthanides and alkaline-earth elements can be made by use of LiCl. The distribution coefficient for thorium is decreased sharply by addition of fluoride to the LiCl, although, the distribution coefficients for the rare earths are affected by only a minor amount [23].

Results of five corrosion tests with exposure time 250 hrs each done with Li,Be,Th,U/F fuel salt containing also Te additions at 720–740°C in the range of U(IV)/U(III) ratios from 1 to 500 demonstrated that high temperature operations are feasible using carefully purified molten salts and loop internals. In these tests device for voltammetric redox potential evaluation was successfully used [24]. The nickel-based alloys selected for testing had the following compositions (in % mass): HN80M-VI (Mo 12, Cr–7.6, Nb 1.5), HN80MTY (Mo–13, Cr–6.8, Al-1.1, Ti–0.9), HN80MTB (Mo 9.4, Cr 7.0, Ti 1.7, W 5.5) and EM 721 (Cr 5.7, Ti 0.17, W 25.2) [23].

After materials exposure in the fuel salt with the [U(IV)]/[U(III)] ratio from 1 to 100 there was revealed no traces of tellurium intergranular cracking on specimens surface for all alloys under study except HN80MTB. Tellurum intergranular cracking was found on tested alloys only after exposure in fuel salt with [U(IV)]/[U(III)] = 500. For each of the tested alloys the intensity of tellurium intergranular cracking was essentially lower in unstressed state than in stress condition. Study on deuterium permeation through nickel-based HN80MTY and EM721 alloys is also carried out. Temperature dependences of deuterium solubility, coefficients of permeability and diffusion in alloys were built. Next Te corrosion test will focus on Li, Be, U/F fuel salt at 750°C.

FHR

The U.S. Department of Energy's (DOE) Office of Advanced Reactor Concepts (ARC) sponsors the U.S. FHR development efforts. Oak Ridge National Laboratory has technical leadership for the program with Idaho National Laboratory performing key fuel qualification and heat exchanger design tasks. During 2011, DOE awarded a significant new university based integrated research program with multiple, interrelated research tasks and a focus on developing a conceptual design for an FHR test reactor. The project is being performed by a team lead by the Massachusetts Institute of Technology along with the University of California at Berkeley and the University of Wisconsin.

In 2012 the ARC FHR development program focused on maturing the design for the Advanced High Temperature Reactor (AHTR). The AHTR is a design concept for a central generating station type [3 400 MW(t)] FHR. The overall goal of the AHTR development program is to demonstrate the technical feasibility of FHRs as low-cost, large-size power producers while maintaining full passive safety. A pre-conceptual design study on a small, modular FHR (SmAHTR) was also completed in late 2010. The AHTR design studies focused on developing a reasonable core and fuel design and placing the proposed core within a power plant.

Development of a fluoride salt component test facility is under progress. The principal activity was construction of a fluoride salt test loop (see Figure 8). Demonstration of wireless (inductive) heating of fuel element surrogates in a salt environment, integrating silicon carbide components into a fluoride salt loop, and development and demonstration of a fluidic diode for liquid fluoride salt application were the main research topics.

A cooperative research program between U.S. Department of Energy and the Czech Republic Ministry of Industry and Trade was also initiated during 2011. The project's technical objective is improving the understanding of the reactivity worth of lithium isotopically selected $2LiF\text{-}BeF_2$ salt. The program involves provision of U.S. produced isotopically separated salt to the Nuclear Research Institute at Řež for testing the salt's reactivity worth at their LR-0 critical facility.

VI. CONCLUSION

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.

Figure 8: ORNL liquid salt test loop [as design drawing (left) and as constructed (right)]

Although the European and USA interests are focused on different baseline concepts (MSFR MOSART and FHR), 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.

A network on MSR R&D has been active in Europe from 2001 with financial support by Euratom. Partners of the MSR PSSC are involved in the Euratom-funded EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor Systems) project and ISTC#1606 and #3749 projects. ISTC has provided another efficient way of collaboration between Russian research organizations, European partners and non-European partners (USA, Japan, Korea, Canada). A complementary ROSATOM programme named MARS (Minor Actinides Recycling in Molten Salt) project between Russian researches organizations is carried out in parallel to Euratom EVOL project.

The common objective is to propose a conceptual design of MSFR by 2013 as the best system configuration – resulting from physical, chemical and material studies – for the reactor core, the reprocessing unit and wastes conditioning. It is intended to deepen the demonstration that the MSFR system can satisfy the goals of Generation IV in terms of sustainability (Th breeder), non proliferation (integrated fuel cycle, multi-recycling of actinides), resource savings (closed Th/U fuel cycle, no uranium enrichment), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner).

ACKNOWLEDGEMENTS

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NOMENCLATURE

AHTR	Advanced High Temperature Reactor
GIF	Generation IV International Forum
LWR	Light Water Reactor
MA	Minor Actinides
MOSART	Molten Salt Actinide Recycler & Transmuter
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
ORNL	Oak Ridge National Laboratory
PSSC	Provisional System Steering Committee
TRU	Transuranic elements
UOX	Uranium Oxide

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THE ROLE OF THE SENIOR INDUSTRY ADVISORY PANEL IN GIF

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ABSTRACT

In 2005, the Senior Industry Advisory Panel (SIAP) of the Generation IV International Forum was created to provide strategic advice to the GIF on developments in the nuclear industry and projections regarding the future, including the views and insights of industry on next generation reactor systems. Based on R&D progress and plans for the six Gen IV systems, SIAP has provided expert advice on future system deployment and needs and progress toward establishing international frameworks for nuclear safety standards and regulations for Gen IV systems. SIAP has convened 10 times since it was created, reviewing one or more systems or crosscutting topics of relevance to the six systems, including the methodologies on risk and safety and economic modelling. The recommendations and observations embodied in the numerous briefings it has provided to the GIF Policy Group have provided valuable insights on industry views of industrial interest, technical viability, economics, licensing, and industrial infrastructure related to next generation reactor technologies. SIAP is comprised of senior industry officials from GIF member countries. More than 30 experts from the electric power industry, industries associated with potential nonelectric applications of nuclear energy, and nuclear suppliers have served on SIAP.

I. INTRODUCTION

Recognizing that it would be important to the future deployment of GIF technologies to incorporate the advice of industry from the earliest phases of GIF research and development, the GIF Policy Group (PG) established the Senior Industry Advisory Panel (SIAP) in 2005. Comprised of senior industry officials representing the electric power industry, designers, nuclear suppliers, and since 2010, industries interested in nonelectric applications of nuclear energy, SIAP's charter is to provide advice on GIF R&D priorities and strategies. Specifically, the SIAP contributes to the discussion of the following:

- Strategic review of R&D progress and plans for individual systems from the industry perspective.
- Strategic review of progress and plans for crosscut R&D.
- Views on system deployment and future nuclear energy fuel cycles.
- Views on international framework for nuclear safety standards and regulations.

Over the eight years SIAP has been in existence it has convened 10 times, usually in conjunction with a GIF PG meeting, providing recommendations, observations, and insights aimed at providing GIF designers and researchers and the governments that are focused on larger policy challenges (security of fuel supply, climate change, waste management) with the perspectives of plant investors whose focus is on risk – that is, the licensing, economic and political risks associated with siting, building and operating nuclear energy technologies.

During its history, SIAP has reviewed the progress of development and nature of collaborative R&D for all six Gen IV systems, conducted reviews related to cross cutting topics of relevance to the six systems, and has reviewed two of the three crosscutting methodologies: risk and safety and economics. Its most recent areas of focus have been on Sodium Fast Reactor safety, including the GIF initiative to develop safety design criteria for sodium fast reactors and the business case for nonelectric applications of Very High Temperature Reactors.

Its recommendations and expert views embodied in the briefings and summarized in this paper have provided the GIF Policy Group with insights on industrial interest, technical viability, economics, and licensing and industrial infrastructure needed for Gen IV technologies. These recommendations have been considered by the GIF Systems Steering Committees, Project Management Boards, and Methodology Working Groups.

II. BACKGROUND

The SIAP serves as a strategically focused "review and advisory" panel to the GIF PG. Currently represented by up to three members per GIF member country, each SIAP member serves for a 3-year renewable term. SIAP brings a needed commercial/industry perspective to GIF R&D collaborations. The aim is to increase the likelihood of successful future deployment by bringing a long-term view to its assessments of specific systems. SIAP advises GIF on priorities, schedules, "market pull" opportunities and implementation possibilities. The SIAP is not, however, a management oversight group or a document review committee.

SIAP nominally meets once per year, in conjunction with one of the two GIF PG meetings. While the early meetings of SIAP were largely organizational in nature, exploring their role and focus, how the PG and SIAP could best interact, and identifying major issues related to commercialization that needed to be explored as the collaborative R&D and technologies matured, over the last several years SIAP had settled into a routine of reviewing specific systems or topics identified in advance by the GIF PG. Through 2012, SIAP has performed reviews of the following topics:

- Considerations for commercialization of Gen IV systems (overarching plan for commercialization, moving toward design convergence, quality management system, adopting a product focus, and developing requirements and specifications for each system that reflects priorities of key stakeholders, etc.).
- SSC Progress and Key Issues (SFR, VHTR, GFR, SCWR).
- Potential for Small Modular Reactors.
- SFR Deployment.
- SFR Safety.
- SFR Safety Design Criteria.
- Business case for nonelectric applications of VHTRs.
- Economic Modelling Life Cycle Cost Methodology.
- Risk and Safety Methodology.
- A follow-up on GIF project management.

III. KEY CONCEPTS/ADVICE FROM SIAP

SIAP has used the analogy of Voyages of Discovery as a way of highlighting some of the key development considerations that need to be satisfied for successful industrial deployment. The overall GIF effort can be likened to a big fleet of many ships all engaged in exploration. Smaller sub-groups of ships, each a separate flotilla, concentrate on exploring specific shorelines, geographic areas or sea passages. These correspond to the work being pursued by the six GIF reactor system concepts. In essence, for a voyage to be successful, several key conditions must be met:

- The fleet, the flotilla and each ship need clear objectives, and clarity about who is in command.
- A documented record is needed of the "track" or route followed. Each ship or system requires an accurate log (record of all experiences and choices made, discoveries, setbacks, damage to equipment, etc.), which is analogous to establishing and maintaining a quality management system. The author or custodian of the logbook is the ship's captain or in R&D parlance, the "design authority". Without the logbook, how can we be sure in future years that the voyage happened or know what was learned?
- Equipment must be tested and crews must be trained and qualified; different stages of the adventure need different skills. In the early phases, GIF System Steering Committees are established with participants who have strong R&D expertise. Later on some people are needed whose strengths are industrial system design and integration.
- There needs to be recognition that ships are independent once out of port and that

there are various stages of discovery as the voyage progresses. GIF systems must adapt to those discoveries as progress is made through the stages of technology readiness.

Finally, as with any voyage of discovery there are both risks – strong currents, adverse winds, disease, wild animals, friendly/ unfriendly peoples – and opportunities – riches. A successful voyage requires an understanding of the risks and how they can be dealt with.

Similarly, in the context of Gen IV development, there are sets of risks to be considered. Investors are primarily focused on licensing, economic and political risk. Governments have to consider security of energy supply, climate change, actinide burning and waste management. To face up to these market realities researchers and designers have to consider aspects such as safety limits and margins, use and failure modes, operability and service lifetime, need for in-service inspection, constructability and manufacturability, necessary underlying infrastructure for transport deployment and operation, decommissioning approach, unique safety and licensing issues, need for unique codes and standards, unique public acceptance issues, monitoring and diagnostics, among others.

The following are key issues that SIAP recommended the GIF pursue:

- *The need for a quality management system to ensure that value is created and captured was the first advice that SIAP gave GIF and one of the first recommendations that GIF tackled. It is a theme that has been revisited and embellished upon by SIAP over the years.* In establishing a quality management system, SIAP recommended the GIF consider questions like: will the work survive a regulatory challenge in 20 years time when the originator is not there to defend it; can a peer researcher independently repeat the work; can a regulator accept the result with confidence? SIAP acknowledged the need for a graded approach to quality assurance commensurate with the phase of the R&D, the need for long-term recordkeeping that would help address issues such as obsolescence, the need for management to set standards and expectations which flow down to the researchers, a minimum program of audits of the GIF research, and an annual review of audit results and actions taken.
- *Moving toward design convergence by establishing pre-conceptual design principles common to designs that allow the designs to advance through R&D and not primarily through an early down-selection.* SIAP also noted the need for a "design authority" that maintains the records of R&D results and rationale for decisions as part of a living business case for each system. Indeed, this is one of the recommendations that the GIF developed a White Paper on implementation that was conveyed to the System Steering Committees.
- *Early regulatory involvement is needed.* SIAP recommended early interactions with regulatory bodies in part to help identify systems or options that are ultimately not licensable; noted the importance of early attention to codes and standards for new technologies as their establishment can be on the critical path for system design and licensing; recommended a simple, high level safety case philosophy that describes the reactor concept, use and failure modes, design basis and beyond design bases events, and why it is safe and noted the need for the safety case philosophy to be understandable and available to the public without the need for translation from complex technical language. Effort also has to go into facilitating the gaining by regulatory staff of the knowledge required to evaluate new technology, which is different to current licenced designs.
- *Six critical success factors for deployment: coherent, visionary, stable leadership; knowledgeable government support and education of key communicators inside government; strategic level communications with the regulator: effective use of experience from previous relevant loops and reactors; ability to substantiate claims under intense scrutiny; definition of critical decision points; and early linking to industry organizations such as WANO, INPO, and EPRI, international organizations like IAEA, and regulatory organizations like MDEP.*
- In their review of *SFR deployment*, SIAP highlighted the importance of maintaining awareness of the market for SFRs and the need for innovation to improve upon past experience with SFRs. SIAP highlighted

the importance of a User Requirements *Specification* as a means of forcing the market to declare their needs and intentions by focusing on the requirements for safety, size/output, schedule, timescales, inspection, maintenance, operations and staffing. SIAP recommended the development of a common international safety vocabulary and engagement of international regulators in the harmonization of safety objectives, and promotion of cross-recognition of codes and standards between regulatory bodies. SIAP again highlighted the importance of quality assurance, maintenance of records, and creation of comprehensive and detailed compilation of issues and lessons learned from past experience with SFRs. Many of these recommendations could be extended to the other Gen IV concepts.

- Finally, SIAP review of *SFR safety* and the GIF Task Force efforts to develop *SFR safety design criteria (SDC)* resulted in a number of comments and observations related to safety philosophy and approach that were considered in the development of the SFR safety design criteria. A member of the SIAP was invited and attended one of the task force meetings in *summer* 2012. The full SIA3P was briefed on the status of the development effort in November 2012. SIAP believes that the development of the SFR SDC is an important step toward development of codes and standards and recognizes that plans have been formulated to engage IAEA on the effort.

IV. SUMMARY

Properly constituted, populated, and briefed, an industry advisory panel such as SIAP can provide valuable guidance for bringing new reactor concepts to the commercial market.

GIF: THE FUTURE GENERATION

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A panel was convened to solicit the perspectives of graduate students engaged in research and development of Generation IV concepts (see Table 1). Art Wharton of Westinghouse Electric Company LLC moderated the discussion and asked a series of questions about the impact GIF has had on their work and interactions with colleagues around the world.

The comments of the panellists demonstrated the need for GIF to improve its external communication, as the work of the GIF and its contributions to the development of advanced nuclear systems are not well known among the United States student population. On the whole, the students that were invited to participate in the panel had to "Google GIF" to find out about it. The students are a representative sample of who the future leaders of the GIF could be one day, and their engagement with GIF at an early time in their career could ensure sustainability of the program's stated objective to operate into the 2030s. After a little bit of background investigation and further thought, the students were able to formulate a wish list of what GIF could or should do to answer their needs as young researchers. This list includes:

- Provide the outside research world with references, a list of publications related to the development of Gen IV systems – as well as a list of research groups and contact details.
- Provide information on the "pros" and "cons" of advanced nuclear systems.
- Set up collaborations between research organisations and universities.
- Provide information related to bench-marking of computer codes.
- Provide information/references on experimental data that can be used with confidence for design and safety studies. An example was given of a "maximum coolant velocity for lead-cooled system to limit corrosion" whose justification cannot be found or traced by the student. This is typically the kind of information that GIF could provide to help R&D activities.
- Provide access to non-proprietary computer codes that run on modern platforms. Many research codes used in universities are either proprietary when used in the frame of contracts with industry, or outdated, with no support from developers who have long retired. The student called for the use of open source codes.
- Provide regular news briefs of recent developments in the field of Generation IV systems which young researchers could read regularly and easily through social media networks or other electronic means.

Table 1: Student panel

Panellist	University	Area of research
Wesley Deason	Oregon State	Fuel design for gas-cooled fast reactor
William Sames	Texas A&M	Nuclear fuel performance in operation and storage, fuel and materials fabrication, and fuel design
Rodolfo Vaghetto	Texas A&M	Experimental and computational investigation of the Performance of reactor cavity cooling systems
Tommy Cisneros	California Berkeley	Design and analysis of fluoride salt cooled high temperature reactors
Staffan Qvist	California Berkeley	Inherent safety of large liquid metal cooled fast reactors and the physics of breed & burn (travelling wave) reactors

GIF STRATEGIC PLANNING

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I. INTRODUCTION

As the world adjusts to the repercussions of the nuclear accident in Japan, many people are asking about the future of the Generation IV International Forum. Gen IV reactors strive to be safer, more economic, and more sustainable than currently installed reactors. These are still valid goals and, in light of public concern over nuclear power, such attributes seem to be even more important today. The question we need to address is what critical activities does the Gen IV program need to undertake and is the pace of development rapid enough, especially over the next ten years, to support commercialization of Gen IV systems in time to meet environmental goals and support energy security?

Answering this question will require introspective assessments and forward-looking strategic planning to chart the course forward. As GIF has now entered its second decade, it is appropriate to review past accomplishments and to assess the strengths and weaknesses of the GIF.

GIF can be thought of as having two integrated parts, the cooperative research and development component and the governance component. During the start-up process, GIF provided an inspiring R&D vision to Gen IV researchers. Subsequent R&D activities, however, focused on a subset of the Gen IV concepts based on member interests. Moreover, the R&D collaboration that materialized, consisting primarily of information sharing, has not stimulated the desired innovation and leaves in question how to accomplish multilateral demonstrations.

The governance structure, which is fully in place after a decade of evolution, is a powerful international legal framework for multinational collaboration. It has proven to be effective for managing the overall effort. While this forward-looking legal framework for nuclear energy R&D collaboration is important, it has yet to be fully exercised. Ongoing Project Arrangements (with the first ones having been initiated in 2007) will continue to be revised when member countries join an existing Project Arrangement. In the future, as GIF member countries identify new research activities for collaboration, it is expected that GIF members will either negotiate revisions to existing Project Arrangements or establish new Project Arrangements. GIF will need to continue to adapt its governance structure to accommodate the evolution of Gen IV technologies and the associated R&D.

II. PROPOSED GIF STRATEGIC PLANNING ACTIVITY

There is broad support within for strategic planning. The planning effort should look forward at least 10 years, with a focus on the activities envisioned over the next 3 to 5 years.

The three elements of the planning exercise are:

- Updating the technology roadmap
- Strengthening R&D collaboration
- Strengthening ties with other international organisations

In May 2012, in Busan South Korea, the Policy Group chartered a strategic planning task force with three sub teams focused on each of the three planning elements.

III. TASK FORCE STRUCTURE

IV. TECHNOLOGY ROADMAP UPDATE

Reexamination of the Technology Roadmap is warranted due to technology advances over the last ten years, emergence of new reactor concepts, and evolving public sentiment towards nuclear energy. Revision of the Technology Roadmap will be in the form of a summary level report focusing specifically on updating the vision of system missions, plans for prototypes, and R&D needs. The update of the roadmap will examine whether there are new or different technical questions that need to be answered in the future, whether to move to demonstration, whether there are new concepts that may be the focus of new collaborative R&D, and whether and how to expand on Gen IV goals given developments over the last few years, including lessons learned from the Fukushima accident. The update will delineate a set of important and challenging goals, activities and projects that would be accomplished in the next decade through GIF. Such projects could involve both the technical and governance components of GIF. Investigation of new reactor concepts could occur within the construct of existing Gen IV system and project arrangements.

The update will build upon the original Technology Roadmap and overlay current technology advances in fuels, materials, modeling, fuel cycle strategies, etc. in an attempt to confirm that GIF is pursuing the optimum set of advanced reactor concepts. This update is timely in addressing global post-Fukushima concerns for installed reactors, new designs, and advanced reactor concepts.

To date, a draft roadmap update has been completed and presented at the GIF Symposium in San Diego, CA. The draft update was compiled by the sub-team secretariat using contributions provided by the System Steering Committees and methodology working group chairs. The draft covers ten year objectives for the six most promising systems and includes examination of current status, major accomplishments since original roadmap, R&D objectives, nuclear safety objectives and milestones. The draft update also covers ten year objectives for the Methodology working groups, including examination of accomplishments since 2002 roadmap, current activities and future efforts. The roadmap update summarizes 5-20 year R&D milestones for each of the six Gen IV systems as described below.

GFR

- Finalization of design of small experimental reactor.
- Fuel development and qualification.
- Robust Severe Accident strategy formulation.

MSR

- Examination of liquid salt properties and behaviour.
- Reprocessing flow sheets and qualification.
- Development of safety approach.
- Decisions on further development of MSR made in 2025-2030.

SCWR

- Materials testing and selection.
- Decision about an SCWR prototype by 2017.
- In-pile, small scale fuel assembly test in 2017 to 2022 timeframe.

LFR

- GIF Collaboration Agreements put in place.
- R&D focus on materials corrosion and safety.
- Lead– and lead-bismuth-cooled experimental reactors built in 2020-2030: ALFRED in Europe, BREST-300 in Russia, Pb-Bi-cooled SVBR-100.

SFR

- Planned start-up in 2014 of several new SFRs, BN-800 in Russia to be followed by other SFRs.
- Completion of detailed design of ASTRID (France), JSFR (Japan), PGSFR (Korea).
- R&D focus on enhanced safety design options and advanced fuels.
- Advanced Gen IV SFRs expected to start operation in 2020-2025.

VHTR

- Start-up of HTR-PM expected around 2017.
- Near term focus on VHTR with outlet temperatures 700-950°C.
- Longer term R&D on advanced materials and fuels to achieve outlet temperatures greater than 1 000°C.

The final roadmap update is planned to be issued in April 2013 and will be presented to the policy group in May 2013. An important consideration will be which of the six systems are considered to be in the viability, performance or demonstration phase.

V. STRENGTHENING R&D COLLABORATION

GIF has an established legal framework for R&D collaborations and the exchange of technical information and data on scientific and technical activities and methods and results of R&D. This sub-team will examine how well the GIF framework facilitates R&D collaborations and the exchange of technical results. The strengths and weaknesses of R&D collaborations and their mechanisms will be examined for a range of GIF projects from the oldest to projects currently under development. Difficulties encountered in establishing new R&D projects will also be assessed.

The availability and use of experimental facilities is an important component of GIF R&D collaborations. This sub-team will also conduct a survey of any new experimental facilities in GIF member countries to determine their age, condition, utilization status and potential for more effective utilization. This information will be provided to the NEA to update their current facility database.

To date, a survey was used to assess what is working well with GIF R&D collaborations and identify areas for improvements. The survey was in the form of a web based questionnaire including multiple choice, short answer questions and open ended sections to provide comments or concerns. Responses were sought from members of GIF Systems and Project Arrangements, System Steering Committees, Project Managements Boards, Experts Group, Policy group and key researchers and scientists (~250 invited to participate). The survey was completed on 17 September 2012 with 108 respondents from a range of programs, countries and functions. A survey report was compiled and the initial analysis of data has been completed.

Most respondents view GIF favourably (82% rate GIF as effective overall) and 60% characterized R&D collaborations as good or excellent. Most participants however saw room for improvement. Some common areas identified for improvement were: R&D coordination; Communication; Management of Projects/Systems; and Funding for and visibility of GIF.

The next step of the sub-team will be to bin identified issues into important and hard to fix, important and easy to fix, less important and hard to fix and less important and easy to fix. Recommendations for improvement will then be developed in January 2013. A final report will be completed in March 2013 and the recommendations will be presented to the PG in May 2013.

VI. STRENGTHENING TIES WITH OTHER ORGANISATIONS

As part of GIF, there has been an ongoing effort to engage international nuclear energy organisations such as the International Atomic Energy Agency (IAEA), OECD Nuclear Energy Agency (NEA), International Framework for Nuclear Energy Co-operation (IFNEC) and the Multinational Design Evaluation Program (MDEP). As part of the planning effort, it is appropriate to examine how ties with these organisations such as through the GIF Policy Group, the Methodology Working Groups (MWGs) and Task Forces (e.g., safety design criteria TF) can be further strengthened to ensure appropriate regulatory standards are in place for Gen IV systems. Development of new collaborations with universities, industries and academic societies will also be examined.

To date, a draft report was completed which examined strengthening ties with IAEA, NEA, MDEP and IFNEC. These international organisations were examined in the context of objectives, content and method of collaboration. Recommendations for strengthening future collaboration were identified for each organisation and development of new collaborations with universities, industry and academic societies were examined. Final recommendations will be presented to the Policy Group in May 2013.

VII. SCHEDULE

The Policy Group approved the GIF strategic planning exercise in Busan in May 2012. Each sub-team has held four conference calls to date and an in person meeting in San Diego California in November 2012. Each sub-team presented their draft reports and discussions items during the GIF symposium in San Diego. Final reports of the sub-teams are expected by April 2013. Final reports and recommendations for improvement will be presented to the Policy Group in Beijing in May 2013.

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The second Generation IV International Forum (GIF) Symposium took place in San Diego, California, USA, on 14-15 November 2012 in conjunction with the Winter Meeting of the American Nuclear Society (ANS). These proceedings present the latest developments in the GIF programme, and as such represent the GIF Annual Report for 2012. They contain the full papers presented during the first day's open sessions as well as updates on developments related to the six GIF systems, the work performed by the horizontal working groups and the "Safety Design Criteria" task force, and the strategic planning activities of the GIF. The symposium was preceded by the ANS President's Special Session which marked the ten-year anniversary of the GIF Technology Roadmap. Former Chairs William Magwood and Jacques Bouchard were honoured during this special session, in the presence of the current GIF Chair Yutaka Sagayama and GIF Vice-Chair Christophe Béhar. A summary of their speeches is reprinted herein, courtesy of the ANS.

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