Nuclear Safety
2011

Experimental Facilities for Sodium Fast Reactor Safety Studies

Task Group on Advanced Reactor Experimental Facilities (TAREF)

NUCLEAR ENERGY AGENCY

Nuclear Safety
NEA/CSNI/R(2010)12

ISBN 978-92-64-99155-2

Experimental Facilities for Sodium Fast Reactor Safety Studies

Task Group on Advanced Reactors Experimental Facilities (TAREF)

NEA No. 6908

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

This work is published on the responsibility of the OECD Secretary-General.
The opinions expressed and arguments employed herein do not necessarily reflect the official views of the Organisation or of the governments of its member countries.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 29 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Corrigenda to OECD publications may be found online at: *www.oecd.org/publishing/corrigenda*.

Cover photos: Experimental fast reactor Joyo (JAEA, Japan); Fast breeder test reactor [FBTR] (IGCAR, India).

FOREWORD

In 2007, the NEA Committee on the Safety of Nuclear Installations (CSNI) completed a study on *Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SFEAR)* which focused on facilities suitable for current and advanced water reactor systems. In a subsequent collective opinion on the subject, the CSNI recommended to conduct a similar exercise for Generation IV reactor designs, aiming to develop a strategy for "better preparing the CSNI to play a role in the planned extension of safety research beyond the needs set by current operating reactors".

In that context, the CSNI established the Task Group on Advanced Reactor Experimental Facilities (TAREF) in 2008 with the objective of providing an overview of facilities suitable for performing safety research relevant to gas-cooled reactors and sodium fast reactors. This report addresses sodium fast reactors.

The findings of the TAREF are expected to trigger internationally funded CSNI projects on relevant safety issues at the key facilities identified. Such CSNI-sponsored projects constitute a means for efficiently obtaining the necessary data through internationally co-ordinated research.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

Background

The Task Group on Advanced Reactor Experimental Facilities (TAREF) was initiated based on discussions held by the NEA Committee on the Safety of Nuclear Installations (CSNI) and the NEA Committee on Nuclear Regulatory Activities (CNRA) during a joint workshop on the Role of Research in a Regulatory Context (RRRC-2, June 2007). Among other topics, the workshop addressed the challenges that the nuclear community will face when performing safety evaluations of advanced reactor designs, the research that may be needed to perform the reviews, and the possible means for jointly conducting this research. In particular, workshop participants discussed research topics relevant to gas-cooled reactors (GCRs) and sodium fast reactors (SFRs), and recommended that the CSNI organise a task group to identify the needed research and to recommend a path forward.

The CSNI initiated TAREF to provide an overview of facilities suitable for carrying out the safety research that was considered necessary for GCRs and SFRs. Other reactor systems could be considered in a subsequent phase.

The TAREF was created in spring 2008 with the following participating countries:

- Canada
- China
- Czech Republic
- Finland
- France
- Germany
- Hungary
- Italy
- Japan
- Republic of Korea
- United States

India provided useful information on its experimental facilities on SFR safety research for the last meeting of the group (in February 2010) and this information was considered for the conclusions of the task.

Approach

The group decided to build on the experience of a similar activity conducted by the CSNI and described in the report entitled *Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SFEAR)*, which focused on facilities suitable for current and advanced water reactor systems. In particular, the SFEAR method was adopted, consisting of first identifying high-priority safety issues that require research, and then categorising the available facilities in terms of their ability to address the safety issues.

At the first TAREF meeting, it was stated that the SFR-related task would need several discussions for elaboration of a PIRT-like approach (PIRT exercise = Phenomena Identification and Ranking Tables) unlike the GCR-related task that could benefit from the already available PIRT compiled in an earlier (United States Nuclear Regulatory Commission) exercise. Hence, the GCR report was issued first in 2009.

As an initial step, the TAREF participants compiled a questionnaire regarding the main technical issues relevant to SFR safety and deserving further R&D work, and the technical infrastructure that is potentially suitable for experimental studies on SFR safety.

Based on the participants' answers to the questionnaire and consistent with the SFEAR report, the Task Group identified the following technical areas to be addressed:

A. Thermo-fluids.

B. Fuel safety.

C. Reactor physics.

D. Severe accidents.

E. Sodium risks.

F. Structural integrity.

G. Other issues.

The technical areas A, B, C and D address phenomena and issues that are specific to the nuclear industry. The other three technical areas E, F, and G address phenomena that are relevant to the nuclear industry, but for which experience may be broader than nuclear.

Other technical areas such as seismic assessment (except for potential consequences on core compaction), instrumentation and control, and human and organisational factors are not treated here, since they are not specific to the nuclear industry and – within the nuclear area – not specific to SFRs. They were addressed only in broad terms in the SFEAR report, which can be consulted for generic information regarding these areas. Other technical areas such as fuel fabrication, fuel handling and irradiated material investigation techniques (as used in hot cell facilities) were not considered as they are more related to operational concerns or not specific to SFRs (hot cells).

For each of the above technical areas (A to G), the task members agreed on a set of safety issues needing research and established a ranking with regard to safety relevance (high, medium, low) and status of knowledge based on the following scale relative to full knowledge: high (100%-75%), medium (75%-25%), low (25%-0%). Only the issues identified as being of high safety relevance and for which the state of knowledge is low or medium were included in the discussion, as these issues would likely warrant further study.

For each of the safety issues, the TAREF members identified the related facilities that were deemed appropriate to address the issue in question, providing relevant information such as operating conditions (in- or out-of-reactor), operating range, description of the test section, type of testing, instrumentation, current status and availability, uniqueness, etc.

Based on the information that was assembled on both safety issues and related facilities, the task members assessed prospects and priorities for SFR safety research and developed recommendations as to priorities and options for the CSNI regarding facility utilisation through programmes that could be pursued internationally. In particular, the group agreed on the main criteria for priority setting, which was based on the following items [high, medium or low (H, M, L) for each item]:

- relevance of the facility to cover a specific issue;
- uniqueness (e.g., one of a kind for in-pile testing);

- availability for a potential programme addressing the issue; three time windows were considered: 0-3 years, 4-8 years, more than 8 years;
- readiness (e.g., staff available to run it);
- operating cost (<0.3; 0.3-1; >1 million USD), or construction cost (<0.5; 0.5-2; >2 million USD).

The group rated those facilities that were costly to either operate or construct as being ranked high in this category as they were more suitable to host a multilateral co-operative programme than facilities of lower cost that could be supported by one country without the need to organise a collaborative programme.

TAREF members who had proposed facilities were requested to characterise their proposed facilities in relation to the above criteria. Based on this, the group's recommendations for the CSNI were developed.

Conclusions and recommendations

1. The TAREF task proved to be a useful exercise for gathering consensus on the technical areas and issues related to the safety of SFR systems, as well as for identifying a number of facilities that are or will become available in OECD member countries for supporting SFR safety research.
2. Due to the specific context of SFR development characterised by a large R&D activity during the period 1970-1995, then followed by a slow-down period and presently restarting in several countries, a large number of facilities operating in the past with sodium as coolant are no longer available (decommissioned, stopped, on standby) or have been converted to water reactor related purposes. This explains why the availability of relevant facilities for all technical areas is limited in the short-term period, and that the decision to restart or to modify some facilities is under consideration for some. This context also led the group to rank the facilities over three time windows, up to the long term (beyond 8 years from now).
3. Based on the responses received, the highest ranked facilities were identified. These facilities are listed in the three tables for the short, medium and long term (see the end of Chapter 4); it should be noted, however, that the presently planned availability of the facilities under construction, to be restarted or refurbished may not be very precise.
4. It is assumed that facilities available in the short term can be used in the medium and long term; similarly, facilities available in the medium term are assumed to be potentially usable in the long term.
5. The group members agreed that for new SFR projects, the most important and top-tier R&D safety needs concern the technical areas with the following priority order:
   - Fuel safety (B) and Severe accidents (D) issues are of prime interest due to the lack of knowledge on new pin design and materials.
   - Thermo-fluids (A) and Reactor physics (C) issues are of second priority as one can live with the current knowledge when considering some margins to cover uncertainties.
   - Sodium risks (E) and Structural integrity (F) issues may be considered with third priority as they are more design dependent.
6. The need for fuel pin irradiation capabilities under representative conditions of fast neutron flux has been identified as a crucial point for addressing safety issues of high priority.

7. In the short term:

- The Indian FBTR fast reactor can be a valuable resource for irradiation of SFR fuel pins and new materials data; the American reactor ACRR (US DOE) would address issues related to fuel safety and severe accidents under specific conditions (provided there is confirmation of its availability for testing in the short term).
- The German KASOLA (KIT) facility would provide data for the thermo-fluids issues in relation with the CFD modelling approaches.
- The Japanese SWAT-1R -3R facility (JAEA) can be appropriate for studying sodium water interaction in steam generator units; the Indian SFTF facility can be valuable for addressing several issues related to sodium fires; the SURTSEY facility (US-DOE) can be relevant to studies on sodium fires and sodium-water interaction in steam generators.

8. In the medium term:

- The JOYO fast neutron reactor (Japan, JAEA) was identified as suitable to address fuel safety issues related to new fuel pin design (fuel pin performance and new materials database under irradiation, margin to fuel melting, impact of use of minor actinides) and some other issues; however, uncertainty still exists as a decision on the possible repair and operating schedule has not yet been taken.
- Severe accident issues can only be addressed in a comprehensive way for the medium-term period and beyond due to the lack of available facilities for simulation of representative transient conditions in short term with irradiated fuel pins. IGR (Kazakh facility used for JAEA programmes) which is addressing fresh fuel (controlled fuel relocation, debris bed formation) may be a suitable solution in the medium term as plans are under consideration for it to handle irradiated fuel. The VULCANO (France, CEA) can also help for severe accident issues, provided it is refurbished for sodium use. The TREAT experimental reactor (US DOE) was also considered in the medium term for its relevance to severe accident issues (past experimental programmes simulating fast power transients) but the restart of the facility has not yet been decided.
- The MASURCA (France, CEA) may be suitable for core physics issues for providing improved nuclear data of core materials (in relation with high burn-up level, use of minor actinides) and associated uncertainties.

9. In the long term:

- The French ASTRID SFR prototype, although at first designed as an industrial prototype to be transposed to a future first of a kind commercial reactor, will offer some irradiation capabilities and may address mainly fuel performance issues (new cladding materials test and impact of minor actinides under fast flux). The JHR material testing reactor (a new French facility under construction, CEA) may address fuel safety issues (new materials database under irradiation, impact of use of minor actinides, slow transients under specific conditions). The availability for first testing is foreseen in 2017-2020.
- The CABRI experimental reactor (operated by the CEA for the IRSN R&D programmes) was recognised by the group members as the most appropriate facility to address irradiated fuel behaviour under incidental and accidental conditions (fuel safety issues such as margins to fuel melting and deterministic pin failure, severe accident issues such as consequences of various accidents leading to fuel melting, with associated consequences and risk of critical events and energy release); the facility may be available for testing from 2020 (after completion of LWR dedicated safety programmes).

- In the case of innovative design for secondary circuits, the LIFUS5 Italian facility (ENEA) would address sodium interaction with alternative coolant species.

10. Relevant CSNI working groups should be encouraged to share modelling information and discuss modelling activities relevant to SFR safety in order to help focus the potential test programmes and/or enhance the data utilisation for model development.

11. An activity in the field of severe accidents and fission product behaviour in an SFR environment should be considered in the Working Group on Analysis and Management of Accidents (WGAMA), which has advanced reactors on its agenda. This activity may consist of an international standard problem regarding SFR safety issues. This activity could help define medium-term initiatives (3-5 years) for an analytical or experimental international programme in specific areas of interest.

12. In a similar way that is intended for gas-cooled reactor fuel, the Working Group on Fuel Safety (WGFS) should consider a work frame (workshop or technical discussion) to address the safety aspects of SFR advanced fuel designs, including SFR fuel safety needs and WGFS initiatives for addressing high-priority issues through possible use of one of the facilities available in the short- to medium-term periods.

13. The CSNI is to maintain an adequate level of exchange with the CNRA regarding needs and initiatives in the SFR safety area.

1. INTRODUCTION

1.1 Background

In June 2007, the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA) held a joint workshop on the Role of Research in a Regulatory Context (RRRC-2), addressing the needs and priorities for nuclear safety and regulatory research. Among others, the workshop discussed high priority safety issues for current plants and for new reactor construction, identifying focus areas for future research. It also considered the challenges that the nuclear community could face in the long term for performing safety evaluations of advanced reactor designs, the possible means for organising and conducting the needed research and for developing the related infrastructure. In the context of advanced reactors, the workshop discussed research topics relevant to gas-cooled and sodium fast reactors and provided a set of recommendations for CSNI initiatives, considering the experience and the good record gained by the CSNI in promoting and managing international safety research projects [1]. In particular, it was recommended that the CSNI develop a strategy and approach for conducting collaborative programmes to support the safety assessment of advanced gas-cooled and sodium fast reactors. The proposed strategy was to define:

- key safety issues as related to specific design concepts;
- issues that will likely require additional research;
- facility infrastructure needed for developing the required data.

The CSNI further discussed these topics at its December 2007 meeting. As stated in the summary record of that meeting, "although the deployment of the Gen-IV systems is not expected in the short term, the CSNI agrees that initiatives should be taken to identify the technical and safety issues that will likely need to be addressed for these systems [2]. To facilitate this effort, the CSNI established a Task Group to provide an overview of facilities suitable for carrying out safety research on gas-cooled and sodium-fast reactors." The task, created in spring 2008 and denominated as the Task Group on Advanced Reactor Experimental Facilities (TAREF), was to focus on gas-cooled reactors (GCR) and sodium fast reactors (SFR). Other reactor systems could be considered in a subsequent phase.

The countries that expressed interest in participating in the task were:

- Canada
- China
- Czech Republic
- Finland
- France
- Germany
- Hungary
- Italy
- Japan
- Republic of Korea
- United States

The TAREF task follows a similar activity conducted by the CSNI and described in the report entitled *Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors* (SFEAR) [3], which was issued in 2007 and which focused on facilities suitable for current and advanced water reactor systems. In a subsequent Collective Opinion Statement on the subject, the CSNI recommended that a similar exercise be conducted for Gen-IV designs, aiming (among other issues) to develop options on how to efficiently obtain the data that are needed, hence "better preparing the CSNI to play a role in the gradual extension of safety research beyond the needs set by currently operating reactors" [4].

As explained in a subsequent section, the present report has been structured in a manner similar to the SFEAR report mentioned above, primarily in that the main focus is on a set of identified safety issues and hence on recommendations for facility utilisation as related to these issues.

At the first Task Group meeting, it was decided that the GCR-related task could be completed at an earlier stage than the SFR task, considering that a significant number of the safety issues had already been compiled by the USNRC [5]. Hence it was decided to produce two separate task reports, i.e., the first one on GCRs, *Experimental Facilities for Gas-cooled Reactor Safety Studies*, and the present one on SFRs.

For the SFR-related task, the safety issues were firstly derived from answers of the TAREF participants to a questionnaire that was sent before the first TAREF meeting.

Dr. Jennifer Uhle of the USNRC and Ms. Joëlle Papin of the French IRSN were elected as task Chairpersons and led the effort for the GCR and the SFR parts of the task respectively.

1.2 Purpose

Advanced reactors incorporate design features, materials and safety provisions that are likely to require exploratory experiments, confirmatory tests and analytical verification. In order to perform this work, adequate infrastructure must be available including facilities, analytical models and expertise. The main purpose of this task is the identification of facilities – as well as recommendations for an optimal development and utilisation of such infrastructure – in order to produce the necessary data in a timely manner as required for safety assessments.

The TAREF objectives are as follows:

1. To provide an overview of existing or planned facilities suitable for safety research investigations relevant to advanced reactors, with focus on GCRs and SFRs.
2. To summarise the Phenomena Identification and Ranking Tables (PIRT) that have already been carried out in the GCR thermo-fluids and fuel areas.
3. To produce a set of relevant safety issues for SFRs in a similar manner to the one used to produce the SFEAR report.[1]
4. To propose recommendations for an efficient utilisation of facilities and resources for meeting short- and long-term safety research priorities.

1. This approach was chosen since a full PIRT for SFRs was not available as for the GCRs and it was not judged necessary for the TAREF purposes.

This activity is considered important for achieving one of the main goals of the CSNI, which, as described in its operating plan, is to help maintain – and if necessary create – the infrastructure and expertise needed to ensure the continued safety of nuclear power production [6]. This goal can be partly achieved by developing a better understanding of the relevant safety issues through co-operative research carried out in specialised facilities and involving internationally recognised experts.

1.3 Scope

The scope of this activity is limited to technical issues and facilities associated with the safety assessment and operation of fast neutron sodium cooled nuclear reactors in the OECD member countries. In the CSNI operating plan perspective, it covers the following main safety issues and topics:

- new concepts of operation;
- new risk perspectives and safety requirements;
- fuel safety;
- new materials and fabrication technologies;
- transparency of the technical basis for safety assessment;
- maintenance of experimental facilities to address emerging safety issues.

Based on the participants' answers to the questionnaire and consistent with the SFEAR report, the Task Group identified the following technical areas to be addressed:

A. thermo-fluids;
B. fuel safety;
C. reactor physics;
D. severe accidents;
E. sodium risks;
F. structural integrity; and
G. other issues.

The technical areas A, B, C and D address phenomena and issues that are specific to the nuclear industry. The other three technical areas E, F and G address phenomena that are relevant to the nuclear industry, but for which experience may be broader than nuclear.

Other technical areas such as seismic assessment (except for potential consequences on core compaction), instrumentation and control and human and organisational factors are not treated here, since they are not specific to the nuclear industry and within the nuclear area, not specific to SFRs. They were addressed only in broad terms in the SFEAR report, which can be consulted for generic information regarding these areas [3].

Other technical areas such as fuel fabrication, fuel handling and irradiated material investigation techniques (as used in hot cell facilities) were not considered as they are more related to operational concerns or not specific to SFRs (hot cells).

1.4 Approach

As an initial step, the TAREF participants compiled the answers to a questionnaire regarding the main technical issues relevant to SFR safety and deserving further R&D work and the technical infrastructure that is potentially suitable for experimental studies on SFR safety. The questionnaire basically addressed the existence of experimental small scale separate effect facilities or large scale integral facilities – or of plans for their construction – as related to areas such as:

- thermal-hydraulics of sodium coolant (including natural convection and computational fluid dynamics (CFD) code validation);
- fuel behaviour;
- severe accidents;
- sodium risks (sodium fires, sodium-water interactions, etc.).

The existence of relevant data and readiness to share them, as well as the willingness to participate in joint international efforts on experimental work were also addressed in the questionnaire. The outcome of the questionnaire was discussed during the first TAREF meeting and served as an initial basis for structuring the task and identifying the areas A to G.

For each of the technical areas A to G, the task members agreed on a set of safety issues needing research and established a ranking with regards to safety relevance (high, medium, low) and status of knowledge based on the following scale relative to full knowledge: high (100%-75%), medium (75%-25%), low (25%-0%). Only the issues identified as being of high safety relevance and for which the state of knowledge is low or medium were included in the discussion, as these issues would likely warrant further study.

For each of the safety issues, the task members identified the related facilities, available or planned, that were deemed appropriate to address the issues in question, providing relevant information such as operating conditions (in- or out-of-reactor), operating range, description of the test section, type of testing, instrumentation, current status and availability, uniqueness, etc.

The group agreed on the main criteria for priority setting, which was based on the following items [high, medium or low (H, M or L) for each item]:

- relevance of the facility to cover a specific issue;
- uniqueness (e.g., one of a kind for in-pile testing);
- availability of a potential programme addressing the issue (0-3 years, 4-8 years, more than 8 years);
- readiness (e.g., staff available to run it);
- operating cost (<0.3; 0.3-1; >1 million USD), or construction cost (<0.5; 0.5-2; >2 million USD).

The task members set up a ranking of the proposed facilities based on the above criteria, and developed recommendations for the CSNI regarding priorities and options for facility utilisation and programmes that could be pursued through international undertakings in the near, medium and long terms. The group rated those facilities that were costly to either operate or construct as being ranked high in this category as they were more suitable to host a multilateral co-operative programme than facilities of lower cost that could be supported by one country without the need to organise a collaborative programme.

1.5 Co-ordination

In assembling the information contained in this report the TAREF participants have the benefit of input from the CSNI working groups (WGAMA on fluid dynamics and accident issues, WGFS on fuel issues and WGIAGE on structural material issues) and from CSNI members. In addition, the CNRA and external organisations (IAEA and others) were offered an opportunity to comment on a draft of the report.

1.6 Organisation of the report

The remainder of this report is organised as follows:

- Chapter 2 provides a short overview of the SFR systems: loop and pool type systems, modular concept, different fuel elements and anticipated design evolution.
- Chapter 3 contains an outline of the seven technical areas and a description of the associated safety issues, explaining the main phenomena involved and the safety implications (in terms of figure of merit). As mentioned earlier, only issues of high importance and low-to-medium knowledge have been considered. Chapter 3 also contains the list of facilities identified for each safety issue.
- Chapter 4 presents the group's conclusions and recommendations regarding CSNI options for facility utilisation, including initiatives for international experimental programmes in support of safety assessments.
- The appendices contain concise facility information provided by the members, the TAREF terms of reference, the group composition and the summary of the four TAREF meetings that were held before issuing the report, relevant to the SFRs.

References

[1] Proceedings of the Joint CSNI-CNRA Workshop on the Role of Research in a Regulatory Context (RRRC2), Paris, France, December 2007, NEA/CSNI/R(2008)3.

[2] Summary Record of the 42nd meeting of the Committee on the Safety of Nuclear Installations (CSNI), December 2007, NEA/SEN/SIN(2008)1.

[3] *Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SFEAR),* NEA/CSNI/R(2007)6.

[4] *CSNI Collective Opinion Statement on Support Facilities for Existing and Advanced Reactors*, NEA/CSNI/R(2008)5.

[5] *Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)* (NUREG/CR-6944).

[6] *CSNI Operating Plan* (2006-2009), NEA/CSNI/R(2007)7.

2. OUTLINE OF SFR SYSTEMS

2.1 Introduction

Fast neutron reactors are characterised by their capability as sustainable energy sources. In particular, sodium fast reactors (SFR) with a closed fuel cycle and potential for minor actinide burning may allow minimisation of volume and heat load of high level waste and provide improved use of natural resources (as compared to only 1% energy recovery in the current once-through fuel cycle).

Among the fast reactor systems, the sodium-cooled reactor has the most comprehensive technological basis as result of the experience gained from worldwide operation of several experimental, prototype and commercial size reactors.

Innovations are needed to further enhance safety, reduce capital cost and improve efficiency and operability, making the Generation IV SFR an attractive option for electricity production.

In the following sections, three SFR concepts are briefly described in terms of their main plant and core characteristics and an outline of the different fuel elements (fuel type, cladding material) is also given.

2.2 Basic description of Generation IV SFR systems – General arrangement

The SFR system 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. The primary system operates at near-atmospheric pressure with typical outlet temperatures of 500-550°C; at these conditions, austenitic and ferritic steel structural materials can be utilised, and a large margin to coolant boiling (about 400°C) is maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. Typical design parameters of the SFR concept being developed in the framework of the Generation IV system arrangement are summarised in Table 1. Plant sizes ranging from small modular systems to large monolithic reactors are considered.

Despite the desirable features of a SFR such as low pressure with excellent heat transfer characteristics, the chemical reactions of sodium with air and water, and opaque characteristics may hamper the deployment of sodium-cooled reactors for electricity generation. The characteristics of sodium also make the in-service inspection and repair (ISI&R) more difficult. Each existing SFR design utilises design measures to increase its reliability in these aspects.

Issues of concern from the viewpoint of SFR safety are the positive sodium void reactivity, except for very small SFR cores, and the possibility of re-criticality in a hypothetical core disruptive accident (HCDA). The effects of these issues need to be evaluated carefully in the safety assessment of the SFR to confirm that the consequences of the accidents are mitigated and retained by the containment functions of the SFR plant and the release of source terms is restricted adequately.

Table 1: Typical design parameters for the Generation IV SFR

Reactor parameters	Reference value
Inlet coolant temperature	400°C
Outlet coolant temperature	500-550°C
Pressure	~ 0.1 to 0.5 MPa
Power rating	50-2 000 MWe
Fuel	Oxide, metal alloy, others
Cladding	Ferritic-martensitic, ODS, others
Average burn-up	150 GWD/MTHM
Breeding ratio	0.5-1.30

The Generation IV SFR R&D focuses on a variety of design innovations for enhanced safety performance (in particular, aiming at a decreased risk of core degradation accidents), actinide management, development of recycled fuels, and improved in-service inspection and repair capability, with a targeted high level of economic performance.

Several sodium-cooled fast reactor conceptual designs have been developed worldwide in advanced reactor development programmes. In particular, the European fast reactor in the European Union, the advanced liquid metal reactor (PRISM) and Integral Fast Reactor Programmes in the United States, and the Demonstration Fast Breeder Reactor in Japan, have been the basis for extensive SFR design studies.

Three reactor concepts are briefly described below. These designs cover a wide range of reactor size and configuration options.

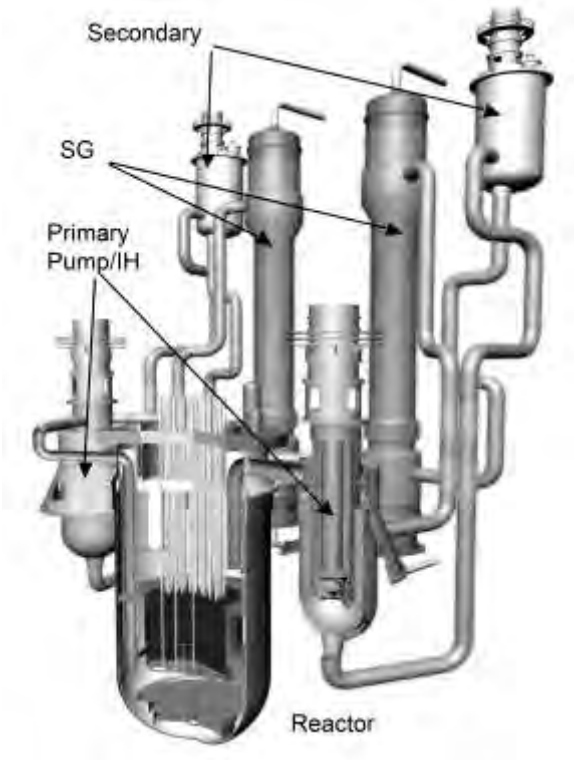
2.2.1 Loop configuration SFR

In the loop configuration SFR, the primary pumps and the intermediate heat exchangers (IHXs) are located outside the reactor vessel (RV) with interconnecting pipings. The characteristics of the loop configuration SFR can be summarised as follows:

- Loop configuration design is able to enhance the reliability of the structures and ease identification of the points to be inspected, since it has just simple internal structures inside the RV and reliable single-piece-forged ring can be applied for the crucial load paths of the RV structure.
- The access to the primary components is easy, since they are separated from the reactor vessel. This feature also provides the flexibility for system modifications and extensive maintenance through the lifetime of the plant.
- The smaller reactor vessel is advantageous for the aseismic design.
- The sodium inventory in the primary system is approximately two to three times smaller than for the pool configuration SFR. This results in a smaller heat capacity and in a faster temperature rise in case of abnormal transients such as the loss of the heat removal system.
- On the other hand, the large difference of vertical elevation between the core and IHX enhances the natural circulation of coolant in well-defined coolant paths by the piping.
- The sodium leakage from the primary components needs to be accommodated so as to prevent a sodium fire and the loss of coolant from the primary system.

A recent concept of the loop configuration SFR is the JAEA sodium fast reactor (JSFR), which is a sodium-cooled, MOX fuelled, advanced loop-type design evolved from Japanese fast reactor technologies. The conceptual plant design is shown in Figure 1.

Figure 1: JAEA sodium-cooled fast reactor (JSFR)



The JSFR design employs several advanced technologies to reduce the construction cost: "compact design of reactor structure", "shortened piping layout", "reduction of loop number", and "integration of components". These measures include innovative technologies such as 12Cr-steel with high strength, an advanced structural design standard at elevated temperature and three-dimensional seismic isolation.

In order to make the most of the advantages of the loop configuration SFR and to resolve its drawbacks, the following design measures are adopted in the JSFR:

- All the primary and secondary cooling pipings, the reactor vessel (RV) and the components are surrounded by guard vessels. This design measure eliminates the accident scenario of loss of coolant and sodium fire in the loop configuration SFR even if coolant leakage due to a failure of a coolant boundary is assumed.
- Enhancing the capability of the natural circulation in the loop configuration SFR, the decay heat removal systems consisting of a direct reactor auxiliary cooling system and two primary reactor auxiliary cooling systems are fully operated by natural circulation only. Each heat removal system is capable of removing the decay heat by itself without other systems and this provides redundancy and diversity with regards to decay heat removal.

Other design features in the JSFR which provide enhanced safety both in the preventive and mitigative aspects are:

- The reactor shutdown system is equipped with a self-actuating shutdown system (SASS) as a third independent shutdown system by passive drop of a backup control rod. SASS utilises the loss of magnetic force at Curie temperature as the actuating mechanism to detect the rise of the outlet coolant temperature in an accidental situation.
- The containment of radioactive materials in HCDA is ensured by the concept of IVR (in-vessel retention) of the accident. IVR is achieved by re-criticality free core concept and cooling and retention of fuel debris in the bottom area of the RV with a specific plate to support the fuel debris. The re-criticality free core concept is based on a successful discharge of molten fuel out of the core during HCDAs by means of an effective core and fuel design consideration, such as a fuel sub-assembly with an inner duct.

2.2.2 Pool configuration SFR: example of a 1 500 MWe reactor concept

The pool concept features nearly all the primary sodium coolant inside a reactor tank. Therefore, this main reactor vessel encloses the primary pumps (PP) and intermediate heat exchangers (IHXs), in addition to the internal structures surrounding the core and devoted either to its feeding and supporting or to separation of the various hydraulic plena.

The topics of interest for a pool-type primary circuit can be perceived as follows:

- There is no relevant accident scenario of loss of primary coolant; the primary sodium inventory is managed by safety provisions (e.g., guard vessel which prevents uncovering of the core).
- The large thermal inertia of the reactor-block contributes to slow down any transient of loss of heat sink by absorbing a large amount of core decay heat. The pump inertia itself can bring a grace period allowing the natural convection of the primary sodium to take place in the main vessel.
- There is no risk of break of the hydraulic loop from the core outlet towards the core inlet.
- A very efficient natural circulation of the primary circuit in case of loss of forced flow mode (e.g., pump trip). The sodium flow can be optimised by a 3-D design of the inner vessel.
- A cold sodium plenum at the pump suction upstream acts as a buffer against either thermal shock or gas entrainment towards the core.
- There is a minimised risk of radioactive sodium fire.
- High mechanical resistance of the primary containment against energetic HCDA.
- Ease of radiation protection under normal operation.

On the other hand, topics related to competitiveness and to flexible operational conditions remain challenging and must be addressed through dedicated R&D:

- Limited access for inspection and repair of the under-sodium internal equipments.
- Seismic behaviour of the sodium free-level and large structures.
- Reactor-block compactness limitation due to integrated large components.

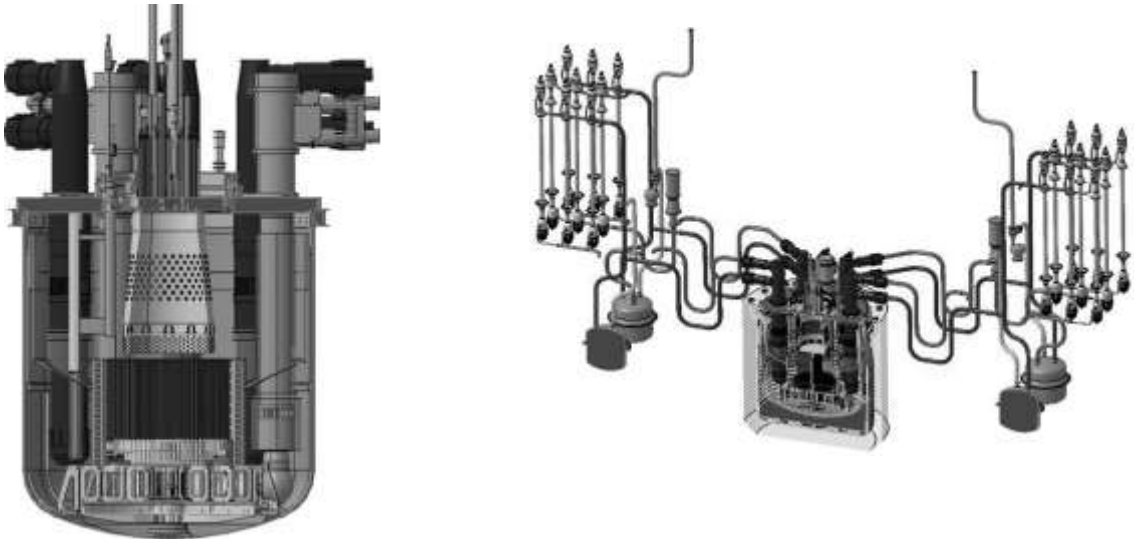
Derived from the European Fast Reactor studies, an innovative preliminary concept of a 1 500 MWe SFR loop reactor has been recently studied within the French cooperative SFR Project (CEA, AREVA and EDF).

The primary loop system described is based on 6 secondary loops, 3 mechanical primary pumps and 6 decay heat removal (DHR) loops (6 time 50% diversified for the half). It integrates new design provisions:

- Ferritic steel is used for intermediate heat exchangers, for an optimal TH design.
- Alveolus roof slab and improved anchorage to the vault. This design allows an optimised manufacturing route of the slab and the improvement of the primary confinement.
- High pressure piping is connected directly to the core diagrid, periphery of which is used as inlet core plenum, in order to eliminate the risk of failure of the connection between pump and diagrid (so called LIPOSO risk).
- The core support strongback is leaning on the vessel bottom.
- Conical inner vessel.
- In sodium fuel handling route.
- Internal and/or external debris tray able to cool and avoid the re-criticality of melted corium.

The main features are illustrated by the 3-D view given in Figure 2.

Figure 2: 1 500 MWe SFR 3-D view of the vessel and general configuration of the modular SGU concept



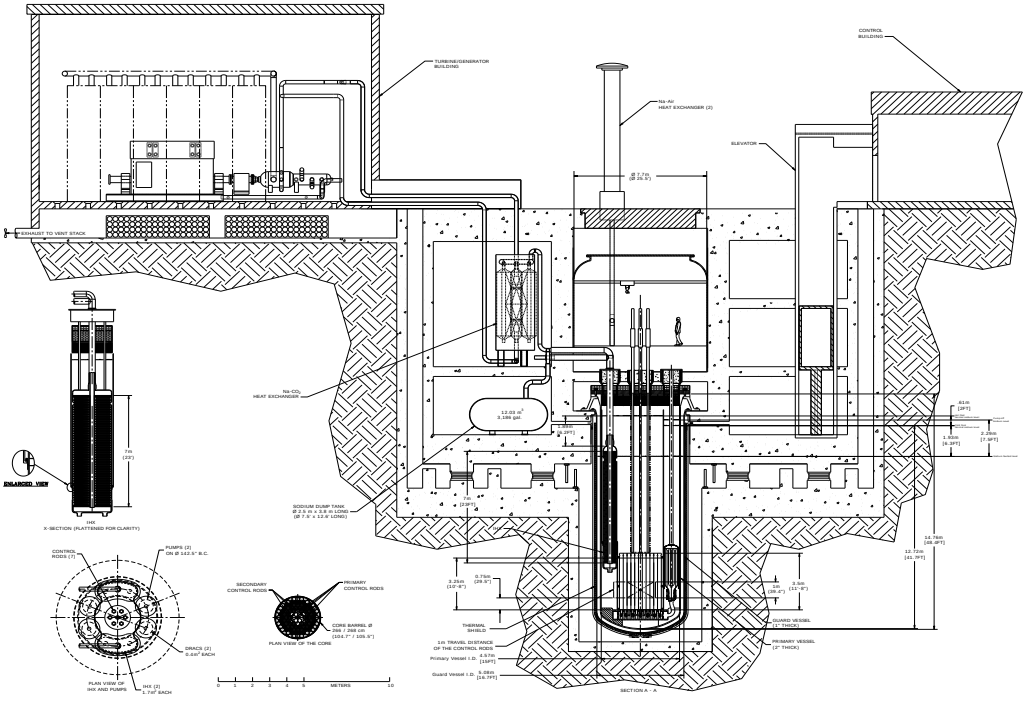
2.2.3 Small modular SFR

The small modular fast reactor (SMFR) aims at exploiting inherent characteristics of fast reactors for application to small grid applications. In a recent United States study, a reactor size of 50 MWe was selected for a specific niche market where industrial infrastructure is not sufficient for larger systems and where the unit cost of electricity generation is very high with conventional technologies. Examples of this situation are remote areas in Alaska, small grid systems in developing countries, and in Pacific-basin islands. The basic goal is to make the operation, safety, and fuel management as

simple as possible for instance, by the use of a long-lived reactor core that eliminates the need for refuelling. The characteristics that enable this approach are:

- The non-corrosive character of sodium coolant does not degrade the reactor core material and primary system components even over very long residence times.
- The excellent neutron economy of fast spectrum and metal fuel can be exploited to design a small core with a conversion ratio near unity, obviating the need for refuelling to account for reactivity losses over an extended cartridge lifetime.
- Innovative design features have been incorporated into the SMFR design including a metallic fuelled core with high internal conversion ratio, inherent passive safety characteristics, simplified reactor configuration for modular construction and transportability, and supercritical CO_2 Brayton cycle power conversion system. The primary and intermediate systems and Brayton power conversion are depicted in Figure 3; the primary and intermediate systems are embedded below the ground level for physical protection. The primary system is configured as a typical pool arrangement with the core, pumps, intermediate heat exchangers, and auxiliary cooling decay heat exchangers all contained within the reactor vessel. The intermediate sodium exits the vessel and flows to the sodium-to-CO_2 heat exchangers.

Figure 3: Elevation view of SMFR system



A key design feature of the SMFR is the long-lived core – 30 years with no refuelling. This long lifetime improves proliferation resistance by eliminating all aspects of on-site fuel management: new fuel acceptance, spent fuel handling, and out-of-reactor storage. The SMFR incorporates all the passive safety features developed for SFR applications to avoid plant damage; this includes a passive decay heat removal system directly from the primary coolant pool.

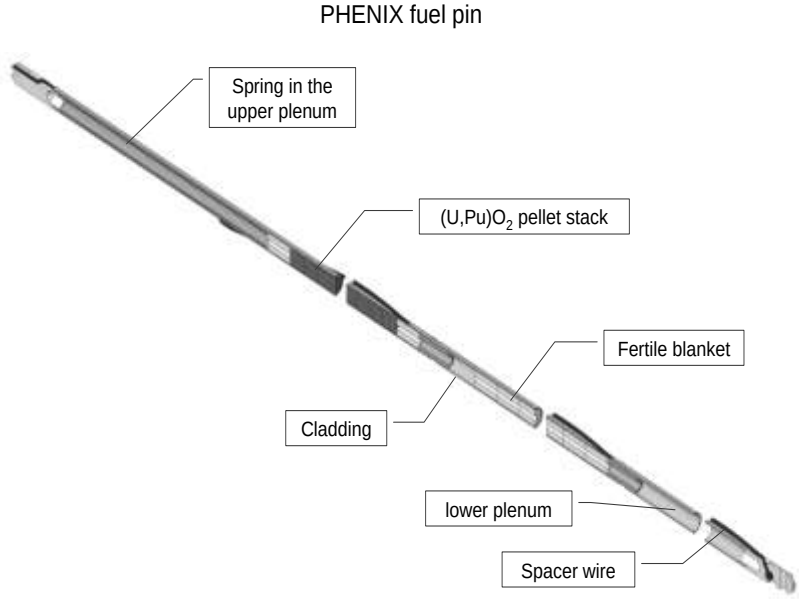
The SMFR utilises a metal fuel form with similar burn-up and fluence limits as employed for the KALIMER design. However, the SMFR operates at a significantly reduced power density to achieve the 30-year lifetime design goal. Thus, the system size is increased compared to a conventional SFR high power density design, and this results in a higher system cost per unit power generation. However, the SMFR energy generation cost is acceptable for the intended niche market application where the small size and design simplicity are more important considerations.

2.3 SFR fuel element characteristics

2.3.1 Oxide fuel

The fuel pin of a fast sodium reactor (SFR) is composed of several elements as illustrated by Phénix fuel in Figure 4.

Figure 4: Example of Phénix fuel pin

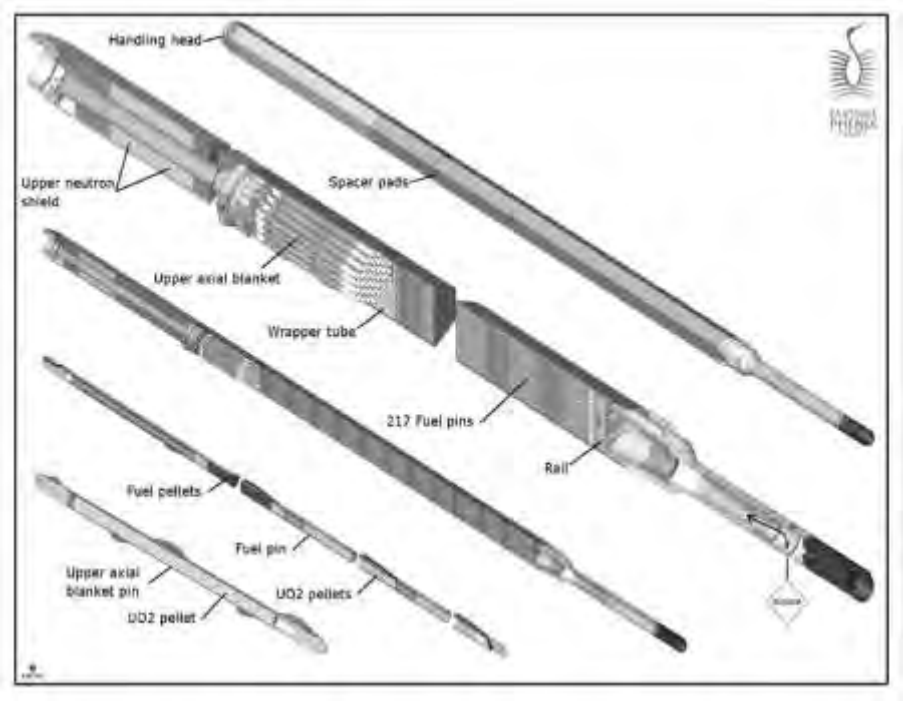


- A cylindrical cladding, closed at both ends by welded plugs, prevents direct contact between fissile material and sodium coolant. The cladding serves as a first safety barrier for the fission product (FP) release and is designed to keep its integrity in nominal and off-normal operations.
- A fissile column that consists of a stack of fuel pellets whose outer diameter is chosen to provide an initial gap (a few hundreds of microns) between the fuel and the internal diameter of the cladding. The most widely-used fuel is the mixed uranium-plutonium oxide: $(U,Pu)O_2$. Depending on the irradiation conditions, pellets are designed without an initial central hole (case of Phénix) or as annular pellets (case of Superphénix).
- One or two axial fertile blankets fabricated with depleted or natural uranium oxide UO_2.

- Two free volumes (gas plenum) located on both sides of the column and initially containing helium gas under 1 bar pressure; the main purpose of gas plena is to accumulate the fission-gas products released during irradiation and to limit the internal pressure below a few dozen bars.
- Other internal structures added such as braces to support the fissile columns and a spring located in the upper plenum to keep the pellet in place before and early in the irradiation.

A helicoidal spacer wire is placed around each pin in order to mix the sodium flow into the pin bundle. Pins are then placed in a hexagonal sub-assembly.

Figure 5: Fuel sub-assembly



To illustrate main SFR oxide fuel characteristics, some figures for Phénix and Superphénix fuel pins are provided in Table 2.

Table 2: Main characteristics for Phénix and Superphénix fast reactor fuel pins

	Phénix standard	Superphénix
Outer cladding diameter (mm)	6.55	8.50
Cladding thickness (mm)	0.45	0.565
Cladding material	CW 15-15 Ti	316 Ti CW 1st and 2nd load 15-15 Ti CW 3rd load
Outer pellet diameter (mm)	5.42	7.14
Inner pellet diameter (mm)	–	2
Plutonium enrichment: Pu/(U + Pu) (%)	20–28*	15–22*
O/M (M = U + Pu)	1.97 ± .02	1.96–2.00
Fuel fabrication density (%TD)**	95.5	95.5
Fuel smear density (%TD)	88.1	82.5
Fissile column length (mm)	850	1 000
Lower fertile column length (mm)	300	300
Upper fertile column length (mm)	–	300
Fuel pin length (mm)	1 793	2 700

* Orders of magnitude for internal and external core.
** TD: theoretical density.

2.3.2 Cladding material

The most frequently used materials for cladding are the austenitic stainless steels which have a very good structural stability in the operating range but are characterised by a fast swelling rate upon irradiation after an incubation period (which depends on the grade considered). The titanium-stabilised cold worked steel such as the "CW 15-15 Ti" grade is today the reference cladding material allowing reaching maximum doses of about 120 to 130 dpa with a cladding swelling below 6%.

2.3.3 Metallic fuel

A typical metallic fuel pin is shown in Figure 6. A solid cylindrical metal fuel slug (as-fabricated piece of metallic fuel) is submerged into liquid sodium and is encapsulated within a cylindrical cladding. The most widely used metallic fuel is a ternary alloy of uranium-plutonium-zirconium (U-Pu-Zr), of which the Zr component is typically 10%. Zr is used as an alloying component because it enhances compatibility with clad and raises the fuel-clad eutectic temperatures.

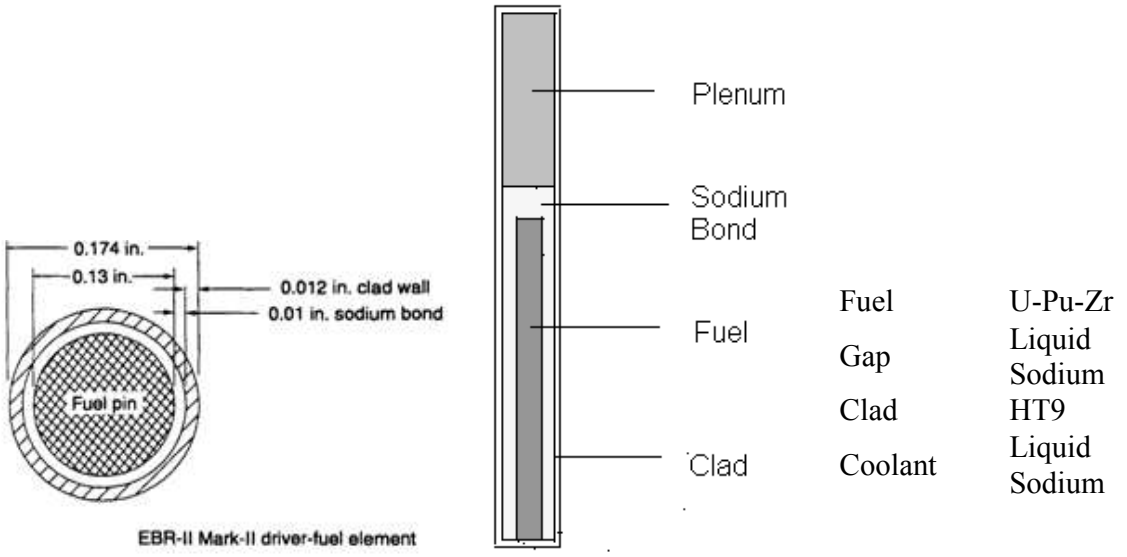
Metal fuel pins are filled with liquid sodium, which provides the thermal bond between the fuel and cladding (compared to inert gas for oxide fuel). This can be done since metallic fuel is not chemically reactive with liquid sodium, and since liquid sodium has high thermal conductivity, this helps to lower the operating temperature for metallic fuel. The typical clad inner diameter would be about the same for both metallic and oxide fuels. However, the outer diameter of the metallic fuel is smaller than that for the oxide fuel, leaving a larger gap between the fuel and clad. Metallic fuel (U-Pu-Zr) can form a

low melting point (675°C) inter-diffusion or eutectic zone when in contact with the cladding, leading to clad penetration and breach. Hence, the metallic fuel pins have a larger gap to prevent fuel/clad contact and subsequent early clad failure in the case of transients, and they are sodium bonded (thermally) for improved thermal conductivity during normal operations resulting in lower fuel centerline temperatures. This is important due to the lower melting point of metallic fuel compared to oxide fuel. There is no central hole in the center of metallic fuel as is sometimes found in certain oxide designs.

The US reference cladding material for metallic fuel for the past 20 years has been HT9, a ferritic-martensitic stainless steel (12Cr-1MoVW) used for its low-swelling characteristics upon irradiation. A certain free volume (plenum) is provided in the upper part of the fuel pin to accommodate the pressure increase due to fission gas release from the fuel. This plenum is filled with an inert gas, typically helium, at atmospheric pressure. The metallic fuel core is about 30% shorter than the oxide fuel core. The smear density for the metallic fuel is 75% as compared to 85% for the oxide fuel. The lower smear density has an interconnected porosity that allows fission gas release to the plenum.

Generally metallic fuel pins do not have springs as the fuel is in the form of slugs and not pellets. However, they can have wire wrap spacers on the outside of the cladding similar to oxide pins. Also, metallic pins may have fertile blankets within the pin at its top and/or bottom similar to oxide pins to enhance breeding.

Figure 6: Description of metallic fuel



2.4 Areas for innovations in SFRs

For the Generation IV sodium fast reactors (SFR) systems, R&D is focused on safety performance, capital cost reduction, efficiency and operability.

2.4.1 Limitation of the sodium-water risk in energy converter systems

One of the main drawbacks to be solved by the standard SFR design is the proper management of the risk of leakage between the intermediate circuit filled with sodium and the energy conversion system using a water Rankine cycle. The limitation or reduction of this risk requires notably an early detection of water leakage to prevent water-sodium reaction propagation and damages (i.e., to other tubes). Two innovative and alternative solutions to this problem can be proposed:

- the replacement of the sodium in the secondary loops by an alternative liquid fluid, less (or not) reactive with water, such as lead-bismuth;
- the replacement of water to another non reactive fluid (i.e., inert gas with a Brayton cycle), that includes the following options (a and b).

a) Some preliminary design calculations have been realised to validate this option using Pb-Bi as an alternative coupling fluid. A first concept of an integrated component – intermediate heat exchanger (IHX) and steam generator units (SGU) in the same tank, thermally coupled with Pb-Bi – has been recently developed, and applied to an hybrid reactor block in which the primary pumps are installed in the reactor vessel (Figure 7) or with a classical loop type reactor (Figure 8).

Figure 7: SFR hybrid concept with IHX/SGU option

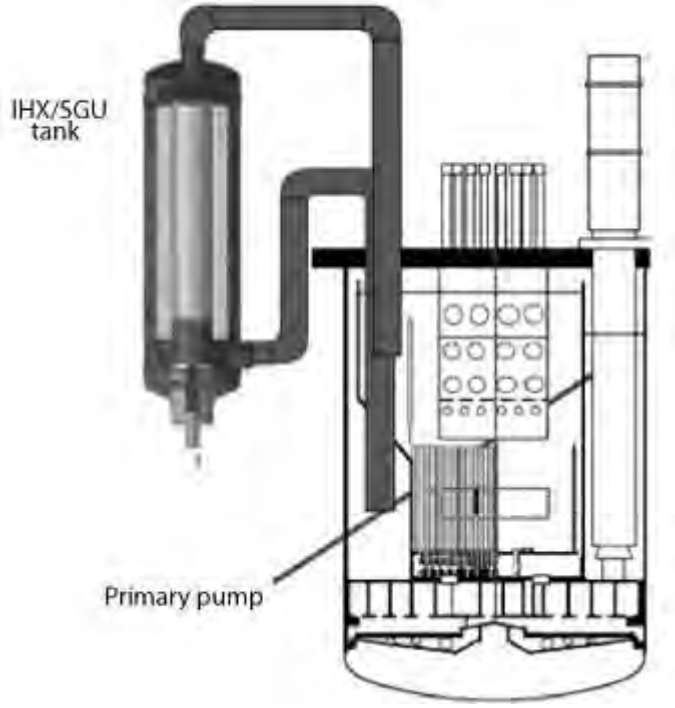


Figure 8: SFR IHX/SGU option: 4 loops and 4 electro magnetic pumps (tertiary circuit is not represented)

Regarding the possible use of Pb/Bi applications, further R&D must address the remaining issues: material corrosion, in service inspection and repair techniques, containment design and costs evaluation.

b) The use of gas energy conversion systems (ECS), may also be considered as an alternative approach to exclude potential sodium-water reactions in ECS.

Many sensitivity studies were performed on various parameters (gases, cycle arrangement, gas pressure, recuperator effectiveness, turbine and compressor efficiencies, core outlet temperature, IHX pinch point, cycle cold point) in order to analyse the conceptual choices.

Gas cycle is a credible alternative to the steam cycles if the safety demonstration can be achieved with a design remaining competitive. The main future R&D works on gas based ECS concern:

- the mastering of the gas insertion risk in the core in the case of suppression of the sodium intermediate loop: the use of an efficient separator must be investigated as a countermeasure system;
- the development of the components technology (compressor);
- the study of the cycle operation and sodium – gas interaction (for SC CO_2: particles formation, dissolution and trapping), efficiency and reliability of the reaction detection systems.

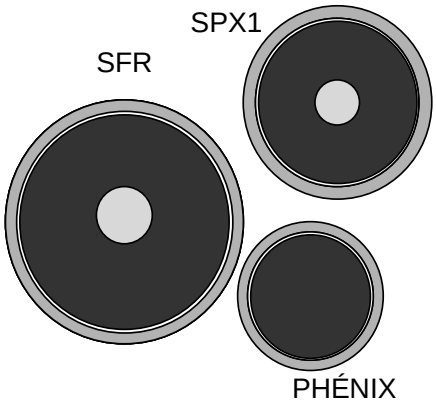
2.4.2 Development of a high performance, safety enhanced core

Advanced core studies for sodium fast reactor are based in order of priority on the improvement of safety with major efforts on the reduction of the core burn up reactivity and coolant void reactivity.

Options for advanced sodium-cooled fast reactors core involve a tight grid pin bundle assembly design associated with large fuel pin diameters, in order to maximise the fuel to sodium ratio (see Figure 9).

Particular dispositions are studied for the core design to prevent the core compaction risk.

Figure 9: Evolution of the pin diameter for Phénix, Superphénix and SFR concept



2.4.3 Development of a flexible fuel, safe and cost effective

As noted above, the most frequently used materials for cladding are the austenitic stainless steels. To reach higher doses and higher fuel burn-up, investigations are focused on ferritic-martensitic steels reinforced by a dispersion of nano-oxides [oxide dispersion strengthened (ODS) steels]. Such materials exhibit a very low sensitivity to swelling as was demonstrated, up to 130 dpa for a classical ferritic-martensitic steel (EM12 grade) irradiated in the Phénix reactor. However, additional R&D is needed for a better understanding of the ODS steel behaviour under irradiation. The optimisation of the fuel with or without minor actinides (MA) is also underway.

Oxide $(U,Pu)O_2$ fuel benefits from the extensive operational and licensing experience. For longer-term applications, SFR core designs with fuel carbide are also evaluated. Regarding oxide fuel, margins on the breeding ratio are important and could be used in order to further reduce the sodium void effect. Moreover, fuel carbide cores require lower heavy metal inventories, which is economically interesting. To further the comparison oxide-carbide fuel, optimisation of the fuel carbide core is needed in considering a new design for the fuel pin and assembly for better use of the carbide fuel specificities.

A large R&D effort is devoted to develop in-service inspection and repair (ISIR) techniques to prevent any risk of core degradation using in-service monitoring, or in-service inspection.

2.5 Safety studies

A robust safety demonstration is a condition for public acceptance of new SFR designs. The deployment of the Gen 3 reactors brought new demanding requirements, with respect to the reactor safety assessment. Large R&D programmes have to be performed to address these issues for SFR both in risk prevention and mitigation of hypothetical core disruptive accident (CDA) in view of avoiding mechanical energy release.

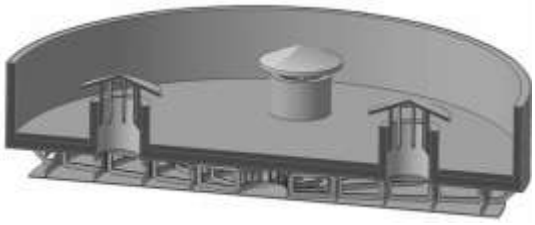
2.5.1 Prevention

- Practical elimination[2] of reactivity accidents:
  - suppression of gas sources or gas trapping area;
  - compactness of core array;
  - core support redundancy; and
  - in-service inspection.
- DHR diversification, efficiency, reliability, passive behaviour.
- Core control:
  - individual computerised SA control;
  - rod withdrawal detection and effects minimisation;
  - early detection of local melts (DND);
  - sodium accidents.
- Sodium leaks: detection, mitigation.
- Sodium/water reactions.

2.5.2 Mitigation

- Provisions for mitigation of core melting risk, and in the case of core melt, preventing energetic criticality sequences.
- Provisions for core melt safe management (core catcher, decay heat removal) (Figure10).
- Defense in depth provisions enhancement, i.e., robust containment against external hazards, in particular:
  - aircraft crash;
  - large seismic event;
  - other malicious attacks.

Figure 10: 1st conceptual design of a SFR core catcher



2. The possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NSG1.10).

2.5.3 Passive safety systems

Passive systems in sodium fast reactors mainly use natural circulation of the coolant and the dissipation of the heat into the environment *via* thermal radiation leading to remove the heat from the core and the reactor vessel.

References

CEA contribution to a SFR Handbook *(to be issued).*

Chang, Y. (2007), "Technical Rationale for Metal Fuel in Fast Reactors", *Nuclear Engineering and Technology*, Vol. 39, No. 3, pp. 161-170.

Crawford, D.C., D.L. Porter and S.L. Hayes (2007), "Fuels for Sodium-Cooled Fast Reactors: US Perspective", *Journal of Nuclear Materials*, 371, pp. 202-231.

François, G., J.P. Serpantié, J.F. Sauvage and P. Lo Pinto (2008), "Sodium Fast Reactor Concepts", Proceedings of ICAPP 08, Anaheim, CA USA, 8-12 June, Paper 8096.

Generation IV International Forum, SFR System Steering Committee (2007), "Generation IV Nuclear Energy Systems – System Research Plan for the Sodium-cooled Fast Reactor rev1.4 Oct.07".

Karahan, A. and J. Buongiorno (2009), "Modeling of Thermo-Mechanical and Irradiation Behaviour of Metallic and Oxide Fuels for Sodium Fast Reactors", MIT-NFC-TR-110, August.

Konishi, K., S. Kubo *et al.* (2006), "The Eagle Project to Eliminate the Recriticality Issue of Fast Reactors; Progress and Results of In-pile Tests", Proceedings of 5th Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS-5).

Kotake, S., T. Mihara *et al.* (2008), "Development of Advanced Loop-Type Fast Reactor in Japan (1): Current Status of JSFR Development", Proc. ICAPP 08, Paper 8226, Anaheim, CA USA, 8-12 June.

Mignot, G., J.C. Klein, M.S. Chenaud, C. Thevenot, A. Ravenet, M. Pelletier, B. Valentin, P. Masoni, P. Dubuisson, S. Delafontaine, L. Nicolas, D. Verrier, A.C. Scholer, D. Ruah, V. Garat, D. Lecarpentier, Ph. Tétart, B. Maliverney, S. Massara *et al.* (2008), "Studies on French SFR Advanced Core Designs", Proceedings of ICAPP 08 Anaheim, CA USA, 8-12 June, Paper 8136.

Nakanishi, S., S. Kubo *et al.* (2008), "Development of Advanced Loop-Type Fast Reactor in Japan (5): Adoption of Self-Actuated Shutdown System to JSFR", Proc. ICAPP 08, Paper 8224, Anaheim, CA USA, 8-12 June.

Uto, N., T. Sakai *et al.* (2009), "Conceptual Design for Japan Sodium-Cooled Fast Reactor (19): Current Status of System Design for JSFR", Proc. ICAPP 09, Paper 9298, Tokyo, Japan, 10-14 May.

Wigeland, R. and J. Cahalan (2009), "Fast Reactor Fuel Type and Reactor Safety Performance", Proceedings of Global 2009, Paris, France, 6-11 September, Paper 9445, pp. 1530-1536.

Yamano, H., S. Kubo *et al.* (2008), "Development of Advanced Loop-Type Fast Reactor in Japan (2): Technological Feasibility of Two-Loop Cooling System in JSFR", Proc. ICAPP 08, Paper 8231, Anaheim, CA USA, 8-12 June.

3. TECHNICAL AREAS, SAFETY ISSUES AND FACILITIES

3.1 Introduction

The worldwide strategy for the future of nuclear fission as discussed inside the Generation IV International Forum (GIF), led to identify the sodium fast reactor (SFR) concept as the most interesting option among the GEN-IV technologies, to potentially address the sustainability issues.

In effect, the SFR with a closed fuel cycle and the potential for minor actinide burning may allow minimisation of volume and of heat load of high level waste and provide improved use of natural resources.

It is also to be noted that the SFR concept has the most comprehensive technological basis as a result of the past experience gained from worldwide operation of several experimental, prototype and commercial size reactors through the period 1970-2000 (for instance Rapsodie, Phénix, Superphénix in France, KNK-II in Germany, Joyo, Monju in Japan, BN-350-600 in Russia, PFR in the United Kingdom, EBR II and FFTF in the United States).

New impetus is now given to the development of SFR technology in several countries through the effective and planned building of reactors from 2010 to around the 2020-2030 period: CEFR in China, ASTRID prototype in France, PBFR in India, JSFR in Japan, BN-800-1200-1800 in Russia, and the advanced burner reactor ABR in the United States.

Future SFRs are now designed with the main objectives of economic competitiveness, increased plant reliability and availability together with an improved safety level as compared to the past (with a requested level at least equal or even higher than GEN-III reactors safety level).

With regard to safety, the specificities of sodium fast reactors and the past experience serve as guidelines to orient the innovative aspects towards reducing the risks of occurrence of:

- prompt critical events and mechanical energy release in the case of severe accidents (risk mainly linked to the positive sodium void reactivity feedback in some core regions, high power density and sensitivity of the core to fuel motion/compaction events);
- sodium fires and chemical interactions with water and air linked to the use of sodium.

The objectives of improved safety together with performance goals will induce evolutions of core characteristics, design strategies to limit accident consequences (prevention, mitigation, passive systems), implementation of high performance materials and new fuel types, etc., and therefore will call for additional knowledge and data in view of development and validation of the analysis tools to be used for a robust safety approach and assessment.

Within the TAREF task, the important phenomena and relevant safety issues have been identified through the PIRT-type approach defined in Chapter 1 (Section 1.4) and recalled hereafter.

As a first step, the completing by the group members of a questionnaire regarding the main technical issues relevant to SFR and deserving further R&D work and technical infrastructure potentially suitable for experimental studies on SFR safety, has been carried out.

Based on the outcome of the questionnaire, the following seven technical areas were considered:

A. thermo-fluids;

B. fuel safety;

C. reactor physics;

D. severe accidents;

E. sodium risks;

F. structural integrity;

G. other issues.

For the SFR, a more global approach than a PIRT has been considered as sufficient and compatible with the time schedule (no PIRT was available as for the gas cooled reactors task). So, for each of these technical areas, the task members agreed on a set of safety issues based on figure of merit (as in PIRT) and needing research and established a ranking with regard to:

- safety relevance (high, medium, low) considering the related figure of merit;
- status of knowledge based on the following scale relative to full knowledge: high (100%-75%), medium (75%-25%), low (25%-0%).

Only the issues identified as being of high safety relevance and for which the state of knowledge is low or medium can be expected to require additional investigation and experimental data to support the safety evaluation of the new SFR projects.

For each of the safety issues and on the basis of information provided by the members, the related facilities (available or planned) that were deemed appropriate to address the issues were identified; the main criteria for priority setting were based on the following items [high, medium or low (H, M or L) for each item]:

- relevance of the facility to cover a specific issue;
- uniqueness (e.g., one of a kind for in-pile testing);
- availability for a potential programme addressing the issue (0-3 years, 4-8 years, more than 8 years);
- readiness (e.g., staff available to run it);
- operating cost (<0.3; 0.3-1; >1 million USD), or construction cost (<0.5; 0.5-2; >2 million USD).

The task members set up a ranking of the proposed facilities based on the above criteria, and developed recommendations for the CSNI regarding priorities and options for facility utilisation and programmes that could be pursued through international undertakings in the near, medium and long terms. The group rated those facilities that were costly to either operate or construct as being ranked high in this category as they were more suitable to host a multilateral co-operative programme than

facilities of lower cost that could be supported by one country without the need to organise a collaborative programme.

3.2 Description of technical areas and related safety issues

3.2.1 Thermo-fluids

Thermo-fluids of the sodium coolant in SFR systems refer to the phenomena associated with sodium flow under forced convection inside the different components of different scales (subchannel, sub-assembly, core, circuits) under all conditions (nominal, incidental and accident) and also to passive cooling of the reactor under natural circulation for residual heat removal. Core thermo-fluids may also concern gas entrainment (for instance from free surfaces) with associated risk of gas bubble passage into the core.

In SFR systems, the sodium flow is characterised by high temperature levels under nominal operating conditions (400°C–550°C at core inlet and outlet respectively), high coolant-structure heat transfer due to large sodium conductivity under liquid phase (especially with the fuel pin cladding), and low Prandlt number.

Inside the core, the sodium flow (forced convection with presence of helicoid wires around fuel pins) must ensure that the maximum allowable cladding temperature is not exceeded.

On the other hand, through the sodium temperature, the thermo-fluid conditions influence the core dynamic system.

So, a precise knowledge of the sodium temperature field inside the reactor core and sub-assemblies is required under operating and transient conditions.

With regard to safety, special attention is given to the prediction of the occurrence of sodium boiling conditions (TNa~ 980°C, 0.2 MPa) that can result from a loss of primary flow and lead to positive reactivity feedback with a subsequent power excursion (unprotected loss of flow=ULOF, potential initiator of a core disruptive accident). Local sodium boiling inside a sub-assembly (due to a local blockage) may also initiate fuel pin degradation (after clad dry-out and melting).

Another safety concern due to high temperature levels of the sodium is the occurrence of temperature fluctuations that may lead to thermal fatigue of the structures through imposed fluctuating stresses in the mixing zones of hot and cold fluid. A correct prediction of these stresses is based on a detailed and reliable evaluation of the temperature variations and of the single phase turbulent flow that needs improvements for low Prandlt number fluids.

Since the start of the SFR studies, sodium thermo-fluid (single and two-phase flow) phenomena have been extensively studied: computational tools have been continuously developed and many experiments have also been performed (CEA, JAEA, FZK,…) that provided the database for heat transfer and pressure drop and related correlations for analysis tools (system, subchannels codes).

Nowadays, advanced computational resources allow more detailed analysis of the thermo-fluid phenomena with multi-scale approaches (subchannel, sub-assembly, system). However, the CFD codes for liquid sodium still need improvement (for instance for turbulence modelling) and data from experiments with high quality instrumentation and measurement techniques at the same scale.

Another important safety issue for the SFR concepts concerns the passive reactor cooling capability by generalised natural circulation under residual power; the safety concern is the prevention of whole-core melting or limitation of the consequences of a core melting accident.

Full scale natural convection tests starting from forced convection at full power and various power levels to natural convection heat removal were conducted at EBR-II (pool type, United States) that have provided measured data and demonstrated that peak coolant and fuel temperatures remain low and do not challenge safety limits. Natural circulation cooling tests were also performed as part of the passive safety programme to demonstrate inherent core cooling capability from refuelling conditions where there is no thermal driving head and from steady state operating conditions. Coolant and fuel temperatures were very low in both cases.

In reactors such as Phénix and Superphénix (pool type, France), the possibility of natural convection has been verified for some circuits but no experimental evidence was available for the whole system: this implied that only computational-based demonstration could be obtained.

So, for the future SFR designs, the evaluation of the efficiency of the implemented passive systems requires an accurate and reliable description of the natural circulation flow regime establishment (time scale) and stability based on knowledge of transition from forced to natural convection.

Detailed descriptions at different scales (component, system) is needed for new fuel assemblies and new reactor designs and local resolution from CFD computations may serve as reference for development/improvement of system scale tools; however lack of data exists for accurate modelling using new approaches (3-D-CFD).

The above considerations led to identify the following major safety issues that would need additional studies:

- flow regime transitions, transport properties, channel flow distribution, sodium boiling (the figure of merits, FoM, being core coolability, sodium boiling onset);
- coolant structure interaction (FoM: structure failure, sodium leak);
- natural convection (FoM: core coolability).

The table below gathers the description of issues and the rationale and ranking levels in terms of safety relevance and status of knowledge following the approach given in Section 3.1.

Safety issue Thermo-fluids	Decription/rationale	Importance, knowledge levels (H, M, L)
A1 Flow regime transitions, transport properties, channel flow distribution, sodium boiling	The issue concerns the sodium flow characteristics and properties under forced convection inside the different reactor components (subchannel, sub-assembly, core, circuits) in all conditions (nominal, incidental and accident). The main concern is the prediction of the sodium temperature field inside the SFR system. Under operating and incidental conditions, the major related concern is the ability to check that the sodium and clad temperatures evolution comply with safety limits; under accident conditions such as loss of flow, the occurrence of sodium boiling is of major importance as it may lead to reactivity insertion and initiate a core disruptive accident. Development of simulation tools is now oriented towards more sophisticated approaches (3-D-CFD codes) that need more detailed data for improved and accurate modelling. The knowledge level was ranked M as there is a lack of knowledge on turbulent flow modelling of low Prandlt number fluid such as sodium (for CFD tools).	**H, M**
A2 Coolant-structure interaction	Fluid structure interaction may influence mechanical loading and behaviour of the core structures. Due to the high temperature of sodium in the upper part of the core with possible high thermal gradients and turbulent flow mixing at high temperature, repeated temperature oscillations may induce thermo-mechanical stresses on the structures and ultimately cause thermal fatigue failure of the material (thermal striping). Moreover, the presence of a spillway may also induce large vibrations on the thermal baffles as shown during the Superphénix commissioning tests. The occurrence of such phenomena is dependent on reactor core design. Prediction of these effects needs detailed evaluation of temperature fluctuations and turbulent flow that can be obtained from CFD approaches and improved modelling; therefore the knowledge level was ranked M.	**H, M**

Safety issue Thermo-fluids	Decription/rationale	Importance, knowledge levels (H, M, L)
A3 Natural convection	The capability of generalised natural convection as a passive reactor cooling system for residual power removal is one of the main issues for improved safety of SFR. The possibility and time scale of initiation of generalised natural convection, its stability, efficiency and reliability (including inter-sub-assembly sodium flow) have to be evaluated, in particular for large cores (and also for the loop type concept). For sub-assemblies with a fairly low power (breeder, fuel sub-assemblies in internal storage positions,…), natural convection inside a sub-assembly is the way to remove power in case of loss of cooling inside the sub-assembly; its efficiency depends on the design, on sub-assembly power and on the power distribution among the pins. Tests have been already performed for a limited number of geometries, and the capability of experimental checking of entire systems has to be examined in the future (design dependent phenomena). A correct and reliable description of the phenomena (at different scales) using simulation tools is needed for new fuel assemblies and new reactor designs. The knowledge level was ranked M as lack of data exists for accurate modelling using new approaches (3-D-CFD).	**H, M**

3.2.2 Fuel safety

This area deals with SFR fuel pin behaviour under operation and under incident/accident transients within the design basis.

During in-reactor operation, fuel pins are subjected to thermo-mechanical and physico-chemical phenomena that lead to structural and mechanical changes in both fuel and cladding materials depending on temperature level and burn-up evolution (i.e., fission gas accumulation and release, cladding embrittlement, pellet-clad gap size, etc.); such changes may jeopardise the ability of the fuel pins to withstand design basis accidents such as slow power transients.

Therefore, detailed knowledge and understanding of the state of the fuel under irradiation is of high importance as this corresponds to the initial fuel conditions at the beginning of an accident (T0-state) and influences the fuel transient behaviour (DBA, severe accidents). This point has been thus retained as a safety issue for all fuel types and burn-up levels, the figure of merit being margin to fuel melting, loss of primary barrier and source term.

With regard to safety under operation and DBA transients, two other generic concerns are the margin to fuel melting and the margin to deterministic pin failure (under typical slow power transients linked to control rod withdrawal); both issues also depend on fuel type and burn-up level.

In effect, the evaluation of the power level at which fuel melting could be initiated (power to melt), is of importance as fuel melting represents a first step of pin degradation and may affect the subsequent

pin mechanical behaviour due to formation of a molten fuel cavity under high pressure (pressure results from the retained fission gases and in case of oxide fuel, from the 10% fuel volume increase at melting). Evaluation of power to melt may serve for defining power level limits.

Of similar high importance are the knowledge and evaluation of the deterministic pin failure level that may occur under slow power transients, in comparison to the operating power as it allows one to quantify the fuel pin margin to failure. Under such transients, clad loading depends on fuel thermal expansion, fission gas induced fuel swelling and molten cavity pressure (if any).

For these two issues, the impact of fuel pellet smear density and geometry and of fission gas retention was underlined from the mixed oxide fuel database at different burn-up levels and is expected under slow power transients as:

- Low smear density allows fission gas induced swelling and fuel creep inside the free volumes that leads to porosity increase and associated thermal conductivity reduction resulting in a low power to melt; this effect is increased with high fission gas retention that might be linked to reactor operation at low power levels.
- In the case of fuel melting and annular pellet fuel, high margin to pin failure is likely, due to internal molten fuel motion; conversely, with solid fuel pellets and high burn-up (12 at%), the margin to deterministic pin failure is reduced (failure due to pellet-clad mechanical interaction, at a level close to the onset of fuel melting).

The mixed oxide fuel type $(U,Pu)O_2$ is the most commonly used in the SFR (Phénix, Superphénix, Joyo, Monju, FFTF, BN-600,…) and is also anticipated at least for the first operating phases of the future industrial prototypes (JSFR, ASTRID, CDFR) however with higher burn-up targets.

SFR mixed oxide fuel is characterised by a high operating temperature linked to a high linear power (range of 400-500 W/cm) and low thermal conductivity (it is to be noted that metallic fuel is characterised by a much lower steady-state operating temperature due to its higher thermal conductivity but the relative margin to melting is similar to oxide with Tnominal/Tmelt around 80%).

Extensive knowledge on mixed oxide fuel has been gained from the irradiation of numerous pins in the framework of the past R&D on SFR and of reactor operation of Phénix, Superphénix, Joyo, Monju, FFTF, etc., and also from past experimental programmes (CABRI, TREAT, EBR II, Joyo). Computational tools have been developed on this basis (i.e., GERMINAL code at CEA, SAS-4A); however, in some conditions, the use of empirical laws prevents the analyst from obtaining a reliable prediction capability, and lack of knowledge exists for the high burn-up range. Additionally, in accordance with the general requirements of robust safety demonstration for the future SFR, the development of more sophisticated simulation tools needs more refined data to ensure accurate modelling.

On the other hand, new fuel and cladding materials and new pin designs are anticipated within the on-going and future development of the GEN-IV SFR concepts that calls for improved safety performances together with economics (reliability, availability of the system) and flexible and robust management of the nuclear materials (waste reduction). Such goals will likely induce new fuel designs (fuel type, cladding materials, pin and sub-assembly geometry, etc.) through the search for:

- core concept with reduced sodium void effect;
- high burn-up fuel capability; this will require new cladding materials with resistance against swelling under irradiation [oxide dispersion strengthened-(ODS)- material anticipated];

- minor actinides recycling (insertion of minor actinides inside fuel being an option);
- new fabrication mode and process within closed fuel cycle objective.

Several fuel types are anticipated for the future: mixed oxide fuel from new fabrication process, ODS cladding, carbide, nitride, advanced burner fuel with MA content, etc.

Obviously, new database and assessments of their behaviour with regard to the identified issues will be needed for a robust safety demonstration.

Another issue concerning absorber rods has been discussed within the group. During in-reactor operation, absorber pins also undergo thermo-mechanical and physico-chemical phenomena that lead to structural and mechanical changes in both absorber and cladding materials; however, these phenomena are different from those occurring in fuel pins.

For some absorber materials, the irradiation time leads to the carburation of the cladding with a risk of clad rupture. The absorber material may fracture with associated risk of fragment release inside the sodium in case of clad failure. Part of the knowledge (medium ranked) has been gained from the irradiation of some absorber pins (boron carbide) but for a limited irradiation time and lack of knowledge was identified for all types of absorber designs (absorber and cladding materials) for long irradiation times as foreseen for the future SFR. However, this issue was ranked medium for safety relevance and thus was not retained.

So, within fuel safety area, the main issues that were retained with high importance level and low to medium knowledge level are the following (FoM=figure of merit):

- fuel pin performance under steady-state conditions (FoM: margin to fuel melting, loss of primary barrier, source term);
- margin to fuel melting (FoM: onset of fuel degradation, extension of molten fuel cavity under accident transients, loss of primary barrier, source term);
- margin for deterministic pin failure (under slow power transients) (FoM: loss of primary barrier, source term);
- new fuel pin designs and materials (fuel, cladding) (FoM: margin to fuel melting, loss of primary barrier, source term);
- use of minor actinides (FoM: reactivity feedback, source term).

It should be noted that some points of the description refer to mixed oxide fuel with stainless steel cladding as the fuel type being the most extensively studied up to the present.

Safety issue Fuel safety	Description/rationale	Importance, knowledge levels (H, M, L)
B1 Fuel pin performance under steady-state conditions (irradiation)	The burn-up increase during in-reactor operation is the major parameter influencing the pin state under irradiation as it leads to the following effects: reduction of the fuel thermal conductivity and of fuel melting temperature, increase of fission gas retention and release rate, evolution of fuel micro-structure (cracking, restructuring, central hole evolution, linked to high operating fuel temperature), evolution of pellet-clad gap thickness and composition (formation of a layer of FP compounds), internal clad corrosion, clad embrittlement and swelling (due to dpa). Therefore, thermo-mechanical impact is expected under transients on margin to onset of fuel melting and on margin to pin failure (deterministic). Reliable prediction capability requires more refined data and modelling of the main phenomena. The knowledge level was ranked M for oxide at moderate burn-up. Lack of data and knowledge concerning the behaviour of oxide fuel with low smear density (annular) beyond 6.4 at% burn-up level (well below the target values for new SFR concepts, ~15 at%) and all types of new fuel pin designs (fuel + clad) to be used and up to the target burn-up values (mixed oxide from new fabrication process, ODS clad, carbide, nitride, metallic fuel, advanced burner fuel with MA content) led to a ranking knowledge level of L.	**H, M** **(MOX moderate burn-up)** **H, L** **(MOX high burn-up and all other fuel types)**
B2 Margin to fuel melting	The "power to melt" depends on fuel thermal conductivity, pellet-clad gap conductance, both affected by burn-up level (cf. issue B1). Influence of pellet design (solid, annular, high or low smear density) has also been evidenced through the past R&D work; in particular, with low smear density irradiated fuel (id annular pellet geometry, 6.4 at%) and under slow power transients, the available data (from IRSN CABRI R&D programmes) indicate that porosity increase resulting from fission gas induced swelling and high temperature fuel creep into the free volumes (leading to central hole closure), induce lower thermal conductivity and thus lower power to melt than originally expected. The knowledge level was ranked M for oxide at moderate level as uncertainty still exists on fuel creep at high temperature, on fission gas induced fuel swelling and on impact of higher fuel burn-ups For other fuel types, the knowledge level was ranked L as fuel pin thermal behaviour under irradiation and slow power transients has to be assessed for all types of new fuel pin designs (fuel+clad) to be used and up to target burn-up values: mixed oxide from new fabrication process, ODS clad, carbide, nitride, metallic fuel, advanced burner fuel with MA content, etc.	**H , M** **(MOX moderate burn-up)** **H, L** **(MOX high burn-up and all other fuel types)**

Safety issue Fuel safety	Description/rationale	Importance, knowledge levels (H, M, L)
B3 Margin for deterministic pin failure (under slow power transients)	The fuel pin thermo-mechanical behaviour depends on fuel and clad materials mechanical properties, temperature and burn-up level. Under slow power transients, clad mechanical loading is mainly due to fuel thermal expansion (linked to fuel temperature increase), fission gas induced swelling (effect of burn-up) and molten cavity pressure after fuel melting onset (if any). With solid pellets (high smear density) and high burn-up level (12 at%), pin failure occurs close to fuel melting onset, leading to a lower margin to failure (Pfail/Pnom ~2). Higher pin failure enthalpy thresholds with annular fuel may be expected under slow power transients due to internal molten fuel motion and result in high margin to deterministic failure (for annular fuel at 6.4 at%, Pfail/Pnom >3). Uncertainty still exists for oxide fuel at high burn-up level. Fuel pin mechanical behaviour under irradiation and slow power transients has to be assessed for all types of new fuel pin designs (fuel+clad) to be used and up to target burn-up values: mixed oxide from new fabrication process, ODS clad, carbide, nitride, metallic fuel, advanced burner fuel with MA content. The same ranking as for the above issues has been given.	**H, M** **(MOX moderate burn-up)** **H, L** **(MOX high burn-up and all other fuel types)**
B4 New fuel pin designs and materials (fuel, cladding)	New fuel and cladding materials and new pin designs are anticipated within the on-going and future development of the GEN-IV SFR concepts that calls for improved safety performances together with economics (reliability, availability of the system) and flexible and robust management of the nuclear materials (waste reduction). The new fuel concepts will need R&D work in order to check their behaviour with regard to safety aspects and to establish a database for safety demonstration and analysis. A comprehensive database, both on fresh and irradiated materials (at different burn-up levels), is needed for simulation tools (among them, fuel and clad thermal properties and mechanical properties at different strain rates).	**H, L**
B5 Use of minor actinides (MA)	MA burning is considered in the future SFR concept as it contributes to the waste reduction. The use of MA in the fuel affects the core characteristics and key safety parameters such as: power density and distribution, sodium void reactivity, decay heat and source term. It may also affect the fuel micro-structure evolution and fuel pin behaviour under irradiation and transient conditions. The expected impact depends on MA content and type. Variability of MA loading from reprocessed LWR fuel may result in a wide range of data needs and the impact of various MA loads within one core needs to be studied since MA content will vary between batch loadings.	**H, L**

Safety issue Fuel safety	Description/rationale	Importance, knowledge levels (H, M, L)
	Very little data are available. R&D work is needed on this topic and new modelling has to be developed in order to check and quantify the impact of MA use with regard to safety aspects, including fuel pin behaviour under irradiation and accident transients.	

3.2.3 Reactor physics

Neutron physics determines the reactor behaviour and core power under normal-operation and dictates its transient behaviour through the reactivity feedback coefficients.

Neutronic response of the core depends on core design and geometry, fuel and clad type, burn-up level and pin and sub-assembly geometry.

For sodium-cooled fast reactors (SFR), one major safety issue is the high sensitivity to the sodium void effect that may induce positive reactivity feedback in some zones of the core (especially in case of large cores). Sodium voiding as a consequence of sodium boiling due to a loss of flow or to a gas bubble passage may therefore result in a prompt critical event with a significant rapid power increase.

Smaller cores or cores with smaller diameters would experience more neutron leakage due to voiding and hence negative reactivity feedback.

On the other hand, in ^{238}U, any increase of fuel temperature caused by incidental/accident transients, results in an increase of absorption cross-sections and leads to a reduction of reactivity (Doppler effect). Fuel axial thermal expansion and radial core expansion (structure dilation, assembly bowing) also induce negative reactivity feedback. Both reactivity feedback effects (Doppler, thermal expansion) depend on the thermal properties of the fuel (mainly) and structure and act as mitigating effects against power transients, although with different time scales, but in most cases, they are not sufficient to compensate for void reactivity in the case of sodium boiling or gas bubble passing.

In some analyses of metallic fuel, it is found that Doppler and thermal expansion feedback may over-ride void coefficient for medium size cores: this has to be confirmed taking into account uncertainty.

For the new SFR core designs, reduction of void effect and optimisation of the feedback coefficients is a top tier objective for improved safety performance with minimisation of risk of reactivity insertion. Taking also into account the goals of flexibility on Pu management, this will call for additional R&D needs as the present status of knowledge is mainly based on oxide and metallic fuel cores studies with moderate burn-up level (maximum 12 at%).

Besides, the development of advanced simulation tools (3-D-kinetics), to be used for refined studies and proper evaluation of safety margins, has underlined the need for improved nuclear data of the core materials including cross-sections and associated uncertainties. In particular, impacts of high burn-up level, Pu vector and of minor actinides have to be considered and evaluated.

Core compaction was also discussed as sodium fast reactors under nominal conditions are not in the most reactive configuration and are therefore highly sensitive to any core compaction effect that can generate a prompt critical event.

Safety issue **Reactor physics**	**Description/rationale**	**Importance, knowledge levels** **(H, M, L)**
C1 Doppler, fuel expansion reactivity feedback	Neutronic response of the core is affected by fuel type, high burn-up level and core design. In response to a fuel temperature increase, Doppler effect, fuel axial thermal expansion and radial core expansion (structure dilatation, assembly bowing) induce negative reactivity feedback. These effects depend on fuel (mainly) and structure thermal properties and act as mitigating effects against power transients, although with different time scales; in most cases, they are not sufficient to compensate void reactivity in case of sodium boiling or gas bubble passing. The knowledge level was ranked H for oxide fuel up to moderate burn-up level (~12 at%) on the basis of the past experience. For higher burn-up fuel and for other fuel types than mixed oxide, the knowledge level was ranked L as lack of data and uncertainty exist, in particular, regarding impact of Pu vector and of minor actinides.	**H, H** **(MOX moderate burn-up)** **H, L** **(MOX high burn-up and all other fuel types)**
C2 Reactivity feedback due to sodium voiding	Sodium cooled reactors are highly sensitive to sodium void effect that may induce positive reactivity feedback in some core zones (especially in case of large cores). Sodium voiding as a consequence of sodium boiling due to a loss of flow or to a gas bubble passage may therefore result in a prompt critical event with significant rapid power increase. The knowledge level was ranked H for oxide fuel cores up to moderate burn-up level (~12 at%) based on past experience. For new core designs with higher oxide burn-up or other fuel types, the ranking level was M due to lack of data.	**H, H** **(MOX moderate burn-up)** **H, M** **(MOX high burn-up and all other fuel types)**

Core compaction can be due to seismic or other events such as rupture of the core support plate (hypothetical scenario considered by JAEA) and is dependent on core design. Consequences of seismic loads are also considered for other types of reactors and generic computational tools and related knowledge are available.

High resistance to core compaction effects has to be considered for the new SFR concepts in the frame of prevention. However, based on these considerations, the issue was ranked H for both safety relevance and status of knowledge and thus was not retained as a safety issue needing additional experimental investigations.

So within this area, the issues of high safety relevance and with low to medium knowledge level are the following (FoM=figure of merit):

- doppler fuel expansion reactivity feedback (FoM: core power);
- reactivity feedback due to sodium voiding (FoM: prompt critical event).

3.2.4 Severe accidents

This area deals with the main accident sequences and related phenomena considered within safety studies.

For SFRs, the generalised core melting accident has been studied for a long time due to its potential for prompt critical event occurrence (linked to the reactivity feedback characteristics) with risk of mechanical energy release to the vessel structure.

Other accident sequences are considered as potential initiators of core melting such as uncontrolled passage of gas bubble possibly leading to a prompt critical event (fast power transient), blockage inside a sub-assembly and slow power transient due to control rod withdrawal.

Detailed knowledge of the involved phenomena during those accident sequences is needed for a quantified evaluation of their consequences. So, a first set of safety issues has been identified according to the accident type:

- unprotected loss of flow (ULOF) with a subsequent transient overpower accident and uncontrolled passage of gas bubbles accident;
- local blockage accident;
- slow power transients (ATWS, anticipated transient without scram).

The unprotected loss of flow accident (ULOF) is considered to be the result of loss of primary pump flow due to several potential initiating events such as an electrical break-down without reactor scram. The first phase (some seconds) leads to sodium flow reduction (kinetics depending on pump inertia characteristics) and to an associated power reduction linked to neutronic feedback; thereafter (some seconds later), the power to flow ratio increases so that the sodium temperature reaches the sodium saturation level (boiling onset). Due to the positive "sodium void effect" in the central core regions, the ULOF leading to sodium boiling and channel voiding may result in a core disruptive accident (CDA) with a primary core power excursion (TOP = transient over power) that initiates generalised core degradation: fuel melting, clad failure and/or melting, fuel ejection into coolant with possible thermodynamic interaction, molten materials motion, fuel dispersal and relocation into the channels and possible mechanical energy release. Beyond the primary excursion, recompaction phenomena, formation of large molten pools with melting of the sub-assembly walls may occur (transition phase) and lead to a secondary power excursion (recriticality events) and expansion phase due to fuel or sodium vapour bubbles with possible significant mechanical energy release to the structures (and potential consequences on sodium spray fire after ejection into the containment vessel).

The past CABRI experimental programmes performed at IRSN/CEA with a large international co-operation provided important knowledge on the phenomena related to primary excursion and valuable data for development of computational tools (SAS-4A among others). The TREAT experiments (US-DOE) also addressed these phenomena.

From the existing database (oxide fuel, moderate burn-up, solid pellet or low smear density fuel), pin failure conditions and fuel relocation mechanisms have been identified, but uncertainty still exists for an accurate prediction and for fuel types other than oxide. Moreover, it was underlined that the mass of ejected fuel out of the fissile zone at the end of the primary excursion was not sufficient to prevent the occurrence of any further critical event and that evolution of the accident towards the transition phase could not be avoided. Lack of knowledge exists concerning the behaviour of large molten pools after the primary excursion (effect of non-condensable gases, fluid dynamics of fuel and steel mixture,

mechanical aspects, etc.). The need for efficient core melt relocation paths (for instance control rod guide tubes) was also underlined in view of possible reduction of recriticality.

Local blockage formation in a fuel assembly due to ingression of some external material into the bundle and failure propagation within the bundle starting from a natural pin failure during steady-state operation are regarded as typical accident initiators (according to JAEA). Rather slow pin failure propagation is expected except for extremely pessimistic assumptions of extensive and rapid blockage formation.

The total inlet blockage (TIB) of a sub-assembly (SA) at nominal power has also been postulated as the initiator for a core melt accident in the frame of EFR (European Fast Reactor) studies. It is considered as the envelope of other types of sub-assembly blockage accidents (local or total but more progressive) leading to sub-assembly melting. Due to the complete loss of flow in the faulty sub-assembly, the usual detection systems are not operating (outlet temperature increase in a SA, delayed neutron detection) so that core power is not shut down early. The accident is characterised by over-heating and melting of the fuel pins, degradation of the SA, wall failure and possible propagation of molten materials into neighbouring SAs and further extension of the melting process. The main safety issue is the risk of propagation of the accident beyond the neighbouring SAs that might lead to critical events and generalised core melting.*

The past SCARABEE experimental programme (IRSN/CEA) with bundle tests and fresh fuel pins, underlined that the consequences of such local accidents depend on the efficiency of the detection systems (a rapid detection after penetration of molten materials in the neighbouring sub-assemblies is needed in order to avoid accident propagation beyond these assemblies and core melting). Modelling of the phenomena has been developed but uncertainty exists on the influence of fission gases on detection capability, on molten pool behaviour (pressurisation by fission gases if tight blockages exist in upper and lower parts) and on fuel ejection out of the sub-assembly (reduction of the SA power).

The unprotected control rod withdrawal accident (CRWA = slow power transient) has been identified as one of the most likely initiating events for a core melt accident (for Superphénix, probability was evaluated to 3.9×10^{-6}/year). It is characterised by a slow power increase (about 1-3%Pn/s) and may lead to partial fuel melting inside the pins surrounding the control rod. In case of an adventitious clad failure of one of those fuel pins (due to initial defect, Pfailure = 1/1 000), molten fuel ejection, even at low melt fraction of 10-20%, and pin-to-pin failure propagation leading to whole sub-assembly degradation might take place and result in core melting extension and critical events. The consequences of such an accident depend on the reactor core size: in the case of a large core, local core effects of overpower may be more pronounced than in a small core in which global overpower can be detected early by usual control systems leading to power shut-down.

Past CABRI programmes underlined that the irradiated fuel pin behaviour has a high influence on the consequences of a CRWA linked to the impact of fission gas retention and fuel creep on power to melt (see B2) in case of low smear density fuel. Possible fuel ejection below 20% melt fraction was also highlighted. Potential consequences may thus be expected from operating conditions at low power that result in a higher fission gas retention for increased fuel creep behaviour, lower fuel conductivity and thus lower power to melt. The present knowledge relates to oxide fuel (low smear density, 6.4 at%)

---

* For Fermi-1, sub-assembly blockage caused by a piece coming loose from the Zr core catcher in the reactor vessel led to a core melt accident on 5 October 1966. Since then, recommendation to design inlet nozzles preventing blockage formation was made (no axial flow).

and uncertainty exists for a reliable description of thermo-mechanical behaviour of the fuel (see B2, B3). Lack of knowledge exists concerning the behaviour of oxide fuel at higher burn-up and of other fuel types and the further consequences of fuel ejection in a sub-assembly (at P>Pn) leading to propagation of the pins degradation, melting extension and critical events.

Another safety issue identified as a common topic to accident sequences concerns the thermo-dynamic interaction that may result from contact of liquid sodium with molten fuel (molten fuel coolant interaction – MFCI). Rapid vaporisation of the sodium, channel or sub-assembly voiding and pressure increase may occur with the potential for mechanical energy release to the structures.

The MFCI energetic impact depends on fuel-sodium thermal exchanges, the fuel enthalpy level, sodium temperature and contact modes. MFCI can result from molten fuel ejection into coolant after clad failure or from melt discharge into sodium after sub-assembly wall failure or from sodium re-entry into the degraded zone (from upper zone or from inter-assembly after hexcan failure). MFCI may also promote molten fuel motion and re-criticality events in case of large molten pools (several assemblies up to whole core size).

Existing data come from several past experimental programmes: CABRI, SCARABEE, CORECT2, SUS, MFTF-B, FARO-TERMOS, THINA. The results suggest that sodium needs to be heated up to near saturation before an energetic event can occur (mild MFCI were obtained in CABRI and SCARABEE tests). MFCI is described in severe accident codes such as SAS-4A, SIMMER and other computational tools using envelop approaches; however uncertainty still exists for an accurate quantification.

Lack of knowledge exists concerning MFCI with fuel types other than oxide and especially with the use of carbide fuel since heat transfer with sodium may be increased due to its higher conductivity and due to the possible chemical reaction with sodium; this may result in more energetic MFCI events.

The long-term behaviour of a core melt accident is also a safety issue. The involved phenomena during the post-accident phase are core material quenching with possible fuel-coolant interaction and formation of a debris bed that influences its cooling capability and stability, in view of the prevention of further re-criticality and of melt-through of the reactor vessel.

Debris bed data are only available from large scale testing in FARO-TERMOS experiments, from D-series in ACRR and from FRAG tests (fragmentation tests with thermite). In relation to the objectives of controlled material relocation through dedicated relocation paths and efficiency of the core catcher (both design dependant topics, issue G1), additional knowledge is needed on material relocation and debris bed formation.

Release of fission products is a general concern for all reactor types as the amount of fission products (FP) and actinides within the fuel is the starting point of the source term analysis and strongly depends on the initial amount of Pu and minor actinides as well as on fuel burn-up and management.

The main phenomena involved are the FP release kinetics (depending on the scenario), chemical reaction with sodium, possible retention of FP by the sodium, aerosol formation with a possible impact of fuel-coolant interaction, and transport. The path way to the containment is design-dependent.

In the past, JAEA has obtained the experimental results of FP release from irradiated high temperature MOX fuel and additional FP release experiments are being planned.

Experience from light water fuel can be used for $(U,Pu)O_2$ fuel but there is lack of knowledge concerning the impact of the use of high burn-up oxide fuel, of minor actinides and of innovative fuels.

So, within the severe accident area, the safety issues considered and related figure of merit are listed below and gathered in the following table:

- unprotected loss of flow with subsequent transient overpower accident and uncontrolled passage of a gas bubble accident (FoM: generalised core melting and degradation, criticality events, mechanical energy release);
- local blockage accident (FoM: propagation of the sub-assembly accident beyond neighbouring sub-assemblies, core melting);
- slow power transients (ATWS) (FoM: core melting, recriticality);
- fuel-coolant interaction (FoM: mechanical energy release);
- post-accident decay heat removal (FoM: core coolability, vessel integrity, critical events);
- release of fission products (source term uncertainty) (FoM: source term, radiological consequences).

Safety issue **Severe accidents**	**Description/rationale**	**Importance, knowledge levels** **(H, M, L)**
D1 Unprotected loss of flow with subsequent transient overpower accident and uncontrolled passage of gas bubble accident	Due to the positive "sodium void effect" in the central core regions, unprotected loss of flow leading to sodium boiling may result in a core disruptive accident (CDA) with a primary core power excursion that initiates core degradation: fuel melting, clad failure and/or melting, fuel ejection into coolant with possible thermodynamical interaction, molten materials motion, fuel dispersal and relocation into the channels and possible mechanical energy release; after this first phase, recompaction phenomena, formation of large molten pools with melting of the sub-assembly walls may occur (transition phase) and lead to a secondary power excursion (re-criticality events) and expansion phase due to fuel or sodium vapour bubble with possible significant mechanical energy release to the structures (with potential consequences on sodium spray fire after ejection into the containment vessel). Uncontrolled passage of a gas bubble may also result in a prompt criticality event and fast power transients with risk of pin degradation and core melting. Pin failure and fuel relocation mechanisms have been identified but uncertainty still exists for accurate prediction. Fuel pin thermo-mechanical behaviour under fast power transients (failure conditions as loss of primary barrier) has to be assessed for all types of fuel pin designs (fuel+clad) to be used and up to target burn-up values: mixed oxide from new fabrication process, ODS clad, carbide, nitride, metallic fuel, advanced burner fuel with MA content, etc. In case of CDA, transition phase related phenomena need additional knowledge for modeling (in particular, effect of non condensable gases and hydrodynamic behaviour of fuel and steel mixture); re-criticality potential should be lowered (efficient core melt relocation paths to be studied) and design prevention measures should be studied.	**H, M** **(oxide fuel, moderate burn-up)** **H, L** **(oxide fuel high burn-up, other fuel types)**

Safety issue Severe accidents	Description/rationale	Importance, knowledge levels (H, M, L)
	The status of knowlegde for the issue was ranked M for oxide fuel at moderate burn-up level and L for higher burn-up and other fuel types.	
D2 Local blockage accident	Local blockage formation in a fuel assembly due to some external material ingression into the bundle and failure propagation within the bundle starting from a natural pin failure during the steady-state operation are regarded as typical accident initiators (according to JAEA). Rather slow pin failure propagation is expected except for extremely pessimistic assumptions of extensive and rapid blockage formation. Local melting accidents are also considered as the result of a postulated total instantaneous blockage (TIB) of a sub-assembly at nominal power (envelop approach of local accidents); consequences are sodium boiling, formation of molten and boiling pools (steel, fuel), sub-assembly wall failure and radial propagation towards the neighbouring sub-assemblies with risk of criticality events (depending on the extension of propagation). Uncertainty exists on the influence of fission gases (the case of irradiated fuel pins) on detection capability and on fuel ejection out of the sub-assembly. Hydrodynamic behaviour of a molten pool of fuel and steel (high pressure build-up) needs to be addressed (link with the transition phase phenomena of a CDA). The issue was ranked L in terms of knowledge level. Efficiency of detection systems via a low response time is a main concern to avoid propagation of the accident and calls for new improvements/developments.	**H, L**
D3 Slow power transients (ATWS)	Slow power transient (such as control rod withdrawal accident, CRWA) may lead to partial fuel melting inside the pins depending on design and assumptions and in case of an adventitious clad failure (due to an initial defect), molten fuel ejection (even at low melt fraction of 10-20%) and pin to pin failure propagation leading to whole sub-assembly degradation might take place and result in core melting extension and criticality events. The present database underlined that the irradiated fuel pin behaviour has a high influence on the consequences of a CRWA linked to the impact of fission gas retention and fuel creep on power to melt (see B2) in case of low smear density fuel and to the possibility of molten fuel ejection below 20% melt fraction; potential consequences may thus be expected from operating conditions at low power (higher fission gas retention for increased fuel creep behaviour, lower fuel conductivity and thus lower power to melt). The present knowledge concerns oxide fuel (low smear density, 6.4 at%) and uncertainty exists for a reliable description of thermo-mechanical behaviour of the pins with higher burn-up and on further consequences of fuel ejection in a sub-assembly (at P>Pn).	**H, M** **(oxide fuel, moderate burn-up)** **H, L** **(oxide fuel high burn-up, other fuel types)**

Safety issue Severe accidents	Description/rationale	Importance, knowledge levels (H, M, L)
	Lack of knowledge concerns other fuel pin designs (fuel+clad) to be used and up to target burn-up values: mixed oxide from new fabrication process, ODS clad, carbide, nitride, metallic fuel, advanced burner fuel with MA content. The ranking level on knowledge level reflects these considerations.	
D4 Fuel coolant interaction	Thermo-dynamic interaction between molten fuel and liquid sodium may lead to rapid sodium vaporisation and pressure build-up with possible mechanical energy release to the structures. MFCI energetic impact depends on fuel-sodium thermal exchanges and mixing modes; potential mechanical energy is increasing with the fuel enthalpy level. MFCI can result from molten fuel ejection after clad failure or after sub-assembly wall failure or from sodium re-entry into the degraded zone (from upper zone or from inter-assembly after hexcan failure). MFCI may also promote molten fuel motion and recriticality events. The past results suggest that sodium needs to be heated up to near saturation before an energetic event can occur. FCI with saturated sodium is expected to be more severe (no data available). MFCI is described in severe accident codes such as SAS-4A, SIMMER and other computational tools using simplified approaches. Uncertainty still exists for an accurate quantification and the ranking level of knowledge was M for oxide fuel. Additional knowledge is needed for other fuel types, and especially in case of the use of carbide fuel as heat exchanges with sodium may be increased due to its higher conductivity and to possible chemical reaction with sodium that may result in more energetic MFCI events. With these considerations, the issue was ranked L as knowledge level.	**H, M** **(oxide fuel)** **H, L** **(other fuel materials , carbide fuel)**
D5 Post-accident decay heat removal	As a consequence of core melt accidents, quenching of core material, debris bed cooling resulting from molten materials relocation inside the vessel and subcriticality of the fuel inventory have to be insured during post-accident phase. The efficiency of a core catcher in the reactor vessel has to be studied (design-dependent). Uncertainty exists on material relocation, quenching, debris bed formation and cooling. The issue was ranked M in terms of knowledge level.	**H, M**
D6 Release of fission products (source term uncertainty)	In case of the loss of structural integrity of the primary system, fission products (FPs) may be released outside the vessel, depending on the fuel type, burn-up level and release scenario. The amount of fission product (FP) and actinides within the fuel is the starting point of a source term analysis and strongly depends on the initial amount of Pu and minor actinides as well as on fuel burn-up and management.	**H, M**

Safety issue Severe accidents	Description/rationale	Importance, knowledge levels (H, M, L)
	FP release depends on their location (gap, inter-granular or intra-granular), porosities, dissolved or precipitated (as metals or ceramics) in fuel grains, on their nature and on release kinetics. Differences are expected depending on fuel type: homogeneous $(U,Pu)O_2$, heterogeneous $(U,Pu)O_2$ (MOX like), innovative concepts. Release kinetics is governed by clad failure occurrence (noble gases, volatile), FP migration to fuel surface (with the impact of temperature, oxygen potential, sodium leaching the pin), and chemical reactions with sodium; a correct prediction of FP speciation is needed. The retention by sodium of some FP with the related effect on the source term has to be considered. FP aerosols can be created from nucleation of FP vapours, precipitation of dissolved FP inside the cold zone of the sodium pool (if any) or by dynamic events such as liquid fuel ejection due to fission gas pressure or fuel-coolant-interaction that can lead to the release of liquid fuel droplets containing their (nearly) full inventory of FP and actinides. Another mechanism is linked to fuel vaporisation. Transport of aerosols particles involving phenomena of nucleation, agglomeration, deposition (Brownian diffusion, impaction, gravitational settling, thermophoresis, diffusiophoresis in case of Na vapour condensation) can be likely studied with the available modeling; the possible formation of organic iodine (from decomposition of NaI) has to be examined. Additional data are needed on the impact of high burn-up fuel, use of minor actinides and new fuel types. The issue was ranked M in terms of knowledge level.	

3.2.5 Sodium risks

The use of sodium as a coolant in a fast reactor necessitates consideration of the risk of sodium leakage through a breach in a pipe or a tank. The high reactivity of sodium with oxygen implies that we need to take into account ignition and combustion of sodium when the metal comes into contact with air. In the same way, its high reactivity with water requires us to consider the sodium-water interaction in steam generators.

The sodium fire scenarios concern mainly the secondary and auxiliary circuits of a SFR. For a reactor design like Superphénix, a leak of primary sodium through the reactor cover into the dome in case of core melting could be envisaged. Fire severity depends mainly on break size and on sodium pressure in the circuit that condition sodium flow rate and fragmentation. The position of the damaged pipe in the premise (in the upper part or near the floor) and the presence of structures (obstacle) in the sodium spill trajectory intervene on spill division. In case of low pressure in the damaged pipe situated close to the floor, liquid sodium burns in pool form. On the contrary, when the pressure is high and with obstacles in the sodium trajectory, the sodium burns in form of droplets in the spray before impacting

the wall or floor where a pool fire could take place. The difference between their combustion kinetics leads to distinguish two types of sodium fire: sodium spray fire and sodium pool fire.

The release of combustion heat causes an increase of pressure and temperatures for gas, walls, structures and components in the leak compartment and eventually in adjacent rooms. Depending on the type of confinement of the fire room (totally closed, compartment with outlets or connected to a ventilation network), sudden cooling of the system (gas, wall) at the end of the fire could provoke a peak of under-pressure and fresh air admission in the compartment. These thermal and mechanical loads could endanger the structural behaviour and hence the containment tightness.

Sodium fire products consist of sodium oxides aerosols that invade the compartment and could damage safety equipment like pumps, exchangers or dampers. These particles also cause loss of visibility in the room (increase of opacity), plug the high efficiency filters situated on the ventilation network and in case of release outdoors, are toxic for humans and environment. In case of a leak of primary sodium, fission products are released in the compartment with sodium fire aerosols and these radioactive products have to be confined in the plant to avoid release of such products to the environment.

Another event that could be of concern for fire scenarios is related to heating of the concrete wall caused by the fire. Above 100°C, the water enclosed in concrete vaporises and is released in the compartment where it could react with sodium oxides (sodium fire aerosols) and eventually with unburned sodium.

When the sodium enters into contact with a concrete surface (floor and/or wall), concrete heating causes water vapour release as described above. In this case, vapour goes through liquid sodium and reacts with it to produce hydrogen. When the oxygen fraction is sufficient, hydrogen burns with the sodium. If oxygen is not sufficient hydrogen could accumulate in a part of the room and risk of explosion must be considered in case of air admission in the room. Moreover, depending on temperatures, solid components of concrete like silica react with sodium and sodium oxides through an exothermal process that damages the concrete surface and could weaken concrete structures.

The general figure of merit related to sodium fires is to maintain the main safety functions and to confine radioactive materials. For that it is necessary to make sure that structures withstand the thermo-mechanical load caused by sodium fire in order to avoid fire propagation and dispersion of aerosols. The aerosol release outside has to remain sufficiently low so as not to cause any damage to the environment.

The above events or phenomena involved in the sodium fire scenario allow us to distinguish six safety issues for this technical area. These issues are listed hereunder with the corresponding figure of merit:

- spray sodium fires (FoM: the thermo-mechanical behaviour of structures, safety elements damage and release of sodium fire aerosols in the plant and outside);
- pool sodium fires (FoM: damage of safety elements and release of sodium fire aerosols in the plant and outside);
- sodium fire aerosol behaviour in the plant and outside the plant (FoM: the human intervention and the safety function losses in plant, and their human health effect outside the plant);
- interaction between sodium fire aerosols and fission products (FoM: the release of radioactive material);

- sodium concrete interaction [FoM: the thermo-mechanical behaviour of structures (containment tightness) and damage of safety elements];
- sodium leak evolution and detection (FoM: one associated with sodium spray and pool fires).

For spray and pool sodium fires, large experimental programmes were performed by IRSN in the past especially in the ESMERALDA large scale facility and by FZK in the FAUNA facility. A wide range of experimental studies using the SAPFIRE facility in JAEA has provided much information for spray sodium fires. IRSN experimental studies concerned the development of extinguishing means for medium and large scale pool fires. The above experiments have enabled the development of computer codes for pool, spray and combined fire in one cell and also in multi-cells configuration, with or without ventilation.

For aerosols produced by a sodium fire, aerosol physical behaviour characterisation, atmospheric dispersion, filtration device development, ventilation driving and equipments failure in the presence of aerosols were investigated in IRSN experimental programmes. A large number of results is available. The results of the FAUNA aerosols loop experiments showed that the size of the particle increases rapidly from 1 to 4 µm due to coalescence (high concentration). The reaction chain starts from oxide to hydroxide and then carbonates, depending on the availability of water and CO_2. Several aerosol filter systems have been investigated in the FAUNA aerosol loop.

IRSN performed experimental programmes with silico-calcareous concrete for characterising physical phenomena. Specific concretes less reactive with sodium and metallic liners were tested. JAEA has experimental data to understand the reaction phenomena of the greywacke-based concrete with sodium. Experimental and computational studies at JAEA are currently in progress to obtain extensive phenomenological information on sodium-concrete reaction and to improve the computational modelling.

Several IRSN tests provided conditions in which low flow rate leaks become larger due to the metallic corrosion by sodium products. Performances of existing detection system were measured.

The main points on which lack of knowledge is identified, are related to:

- Sodium spray fires with lower initial sodium temperatures (200-250°C): in these conditions the sodium jet ignition delay is longer than for higher temperatures and experimental data on combustion kinetics during the first phase of such a fire are needed.
- If a sodium fire in a room leads to wall temperatures equal to or higher than 100°C for a sufficient time, water vapour is released and reacts with sodium fire aerosols (sodium oxides) to produce sodium hydroxide and, in some conditions, with liquid sodium to form hydrogen. In case of local accumulation of hydrogen in a room and of oxygen admission, a risk of explosion exists. The release of water vapour from concrete is relatively well known but specific conditions that lead to hydrogen formation and accumulation are to be determined.
- When sodium fire aerosols are released outside the plant, they disperse in the atmosphere and depending on meteorological conditions, they react with humidity and carbon dioxide in the air. Chemical kinetics in these conditions need to be investigated because sodium carbonate and sodium hydroxide do not have the same impact on human health.
- Data related to interaction between sodium aerosols and fission products are needed to allow for consequence evaluations.
- Development of efficient sodium leak detector systems is needed and solutions to limit sodium corrosion have to be investigated. This point is strongly dependent on design options.

From all these considerations, importance and knowledge rankings are made for each safety issue. Among the six issues related to sodium fires, two of them are not kept in the table hereafter:

- sodium pool fires;
- interaction between sodium fire aerosols and fission products.

For sodium pool fires, the importance is high but the existing knowledge is also high. For this two reasons are given: firstly, ventilation driving and extinguishing means have been developed for pool fires in the past and could be used; secondly, future design options will be developed in such a way that large sodium pool fires would be avoided. Therefore the lack of detailed knowledge that remains for flame spreading on a large pool surface especially at low temperature, is not taken into account.

The importance related to the interaction between sodium fire aerosols and fission products is assessed to be medium while its knowledge level is low.

Besides issues related to sodium fire scenarios which are detailed above, the sodium risks area involves issues related to sodium-water reaction:

- sodium-water interaction in steam generators;
- sodium-water interaction in air.

Each of these two issues corresponds to a scenario that is described here under.

Risk of sodium-water interaction concerns sodium of the secondary circuit and water of water/vapour circuit in the steam generator where the two fluids circulate inside and outside steel tubes to exchange heat. Due to the large pressure difference between the two circuits, a tightness default in a steel tube causes the penetration of water vapour into the sodium that provokes a sodium-water reaction. The effects of such a reaction are the following:

- Global corrosion by hydroxide and oxygen that are dispersed in the sodium.
- Local wastage of the leak orifice ("self wastage") and of near structures ("target wastage", steam generator neighbouring tubes or outer steel jacket of steam generator).
- For large leak flow rates (>100 g/s) overpressure and a pressure wave due to hydrogen expansion take place and could propagate in the secondary loop, in particular in the intermediate heat exchanger tubes.
- Overpressure in the secondary loop initiates sodium movements between expansion volumes of the loop.
- Sodium-water reaction is an exothermic process; in case of large leak flow rates (>100 g/s), reaction products are very hot and the temperature of neighbouring tubes increases significantly; these heated tubes subjected to the water/steam pressure swell and could burst with the creation of a large break.

For this issue, the figure of merit is related to the confinement of radioactive materials (primary sodium) in case of damage of the second safety barrier (intermediate heat exchanger tubes).

CEA experimental programmes provide substantial results concerning leak flow rate evolution, pressure waves propagation and mass transfer within the secondary circuit, damages caused to the

neighbouring exchange tubes and problems (efficiency and rapidity) arising from the sodium-water reaction detection.

Experimental and analytical studies were carried out using the SWAT facility in JAEA, concerning self-wastage, target-wastage and overheating of neighbouring heat transfer tubes.

Several computer codes were developed, each of them is dedicated to a specific physical phenomenon like water flow rate evolution, failure propagation, pressure rise within the secondary circuit, overheating behaviour.

Most of these studies were performed with austenitic materials. In the case of ferritic or other steel type use, knowledge will be needed on the phenomena involved and behaviour of the material in case of sodium-water reaction. Consequently all the considerations about this issue are strongly dependent on the design option. Moreover, improvement of the hydrogen detection system is required.

Sodium-water reaction in air is envisaged when two leaks – water and sodium – intervene in the same premise due to an external accident event that causes rupture of both sodium and water circuits of a steam generator or due to an internal accident event like large sodium-water reaction in a steam generator that damages the steam generator external jacket. This reaction could be used for the washing of components during maintenance operations during the reactor life or in the frame of reactor dismantling.

This reaction is different from the sodium-water reaction in steam generators. In particular, oxygen reacts with sodium and hydrogen and the conditions in which products come into contact and amount of products involved are very different. The risk of explosion is to be considered: explosion of hydrogen in the presence of air and also thermal explosion (fast vaporisation) of water in contact with hot sodium. Development of the process involves numerous complex phenomena and their interaction: pressure peaks, gaseous bubble growth, combustion, explosion, sodium fragmentation, etc. Hydrogen detonation is very difficult to achieve, most events are fast deflagrations.

The figure of merit is the thermo-mechanical behaviour of structures.

Some analytical tests were performed by CEA in the past. Computer codes were developed by CEA. Numerous additional investigations are needed on the main phenomena.

Like the previous issue, this one is dependent on the design options.

The last issue concerning sodium risks is totally design-dependent: if another coolant is chosen in place of water, interaction between the new coolant and sodium has to be investigated, that means large R&D programmes have to be carried out. Among the possible coolants, we can find supercritical carbon dioxide, lead-bismuth eutectic (LBE) and lead.

As for sodium-water reactions, the figure of merit is related to the confinement of radioactive materials (primary sodium) in case of damage of the second safety barrier (intermediate heat exchanger tubes).

In the table hereunder, the safety issues are listed with a brief description (second column). In the same column, arguments for ranking related to importance of the issue (H, M, L) and to existing knowledge are given.

Safety issue Sodium risks	Description/rationale	Importance and knowledge level (H, M, L)
E1 Spray sodium fires	Spray fire happens in the case of a large sodium leak on the SFR circuit in a non-inert atmosphere. The consequences of which depend on leak size, geometry and on the pressure in the circuit. The sprayed sodium metal ignites and burns during its path in the air before impacting room walls or floor. Combustion kinetics are fast and the consequences consist in pressure and temperature rises in the leak compartment. The hazard is related to the structures thermo-mechanical behaviour to maintain containment integrity and tightness. Containment, ventilation and filtration systems are designed to withstand these consequences and to confine the aerosols. For mitigating spray fire consequences, parcelling of premises could be used. The database of the experiments mentioned above have enabled computer code development for spray fires, combined spray and pool fires in one cell and also in multi-cells configuration, with or without ventilation. However, some questions remain concerning lower temperature (200-250°C) sodium sprays and the reactions between water vapour released by heated concrete walls and sprayed sodium.	**H, M**
E2 Pool sodium fires	Issue not retained as large pool sodium fires are supposed to be avoided in future SFR designs.	**H, H**
E3 Sodium fire aerosols behaviour	Sodium fires produce sodium monoxide and peroxide aerosols that react with water vapour producing sodium hydroxide particles. Generally aerosol concentration is very high and inhibits human intervention in the leak compartment and adjacent rooms. The concentration and size of aerosols is dependent on time, fire size and humidity. Aerosols may damage electronic equipments. If burned sodium contains radioactive products, sodium fire aerosols could act as a vehicle for transportation and contamination. In case of sodium fire aerosol release outside the building to the environment, sodium hydroxide and sodium carbonate are not equal regarding the toxicity and effect on human health. Atmospheric dispersion has to be considered for aerosol size and concentration determination. Computer codes exist at IRSN and JAEA for simulating sodium fire aerosol behaviour in the room, in the ventilation network and in open atmosphere in case of release to the environment. Complementary experimental data, code validation and modelling improvements are needed related to: • sodium fire aerosol production; • deposition rate on complex surfaces of structures; • the effect of gas flow in case of a spray fire; • chemical composition evolution of aerosols outside the building as a function of meteorological conditions.	**H, M**

Safety issue Sodium risks	Description/rationale	Importance and knowledge level (H, M, L)
E4 Sodium-concrete reaction	The direct contact between concrete and hot sodium causes steam and hydrogen release due to reaction between water vapour released by heated concrete and sodium. The solid material could be involved in exothermal reactions with sodium. The consequences are pressure and temperature increases in the room and explosion risk. Protection devices are designed for protecting concrete surfaces but it seems more difficult to totally avoid water vapour release from concrete and consequently hydrogen formation. Computer codes were developed and were validated on the basis of available experimental data. For some of them, validation has to be completed including modelling improvements (sodium-concrete thermal exchange coefficient calculation). Need for knowledge may arise if a different concrete (other than silico-calcareous or greywacke-based concrete) is used.	**H, M**
E5 Sodium leak evolution and detection	The case of a sodium leak with a low flow rate (<1 cm^3/min) on a pipe or a tank that are thermally insulated is particular: sodium accumulates in the insulated material with possible metallic corrosion, and the sodium spreading is limited so that flow rate could increase notably before leak detection. The flow rate evolution is to be known and efficient detection systems have to be installed to prevent sodium release outside of the thermal insulating material and sodium fire. Technical solution could be investigated to avoid or to limit steel corrosion. Need of development of a more efficient detector system exists. This issue is dependent on the design options.	**H, M**
E6 Sodium-water interaction in steam generators (SGU)	The loss of tightness in steel walls between sodium and water in a steam generator causes vapour penetration in sodium and sodium-water reactions. The damages could be significant and could affect the safety function that consists in confining radioactive materials as the sodium-water reaction propagates in the secondary circuit and may cause damage to the exchange tubes of the intermediate heat exchanger (IHX) which are a part of the second safety barrier. Leak detection systems are required and have to be developed and qualified. Some lack of knowledge and need for improvement remain especially concerning hydrogen detection systems. The physical phenomena involved and the associated lack of knowledge will be strongly dependent on design options: design of vapour generator or heat exchanger, metallic material (for example ferritic steel type material used as material for SGU shells and tubes).	**H, H(1) and L(2)**

(1) Austenitic steel.
(2) Ferritic steel and other steel.

Safety issue Sodium risks	Description/rationale	Importance and knowledge level (H, M, L)
E7 Sodium-water interaction in air	In case of simultaneous sodium leak and water leak in a room or in case of a sodium treatment process using water, a sodium-water reaction in air has to be considered. Phenomena are extremely complex and their consequences could be severe: detonation or deflagration associated with shock waves. Complementary studies are needed on main phenomena (mechanical, thermo-hydraulic, chemical, acoustical) contribution and on sodium dispersion due to water vaporisation and thermal explosion. Strongly dependent on design options	**H, M**
E8 Sodium interaction with alternative coolant species	Reactions have to be investigated: supercritical CO_2, LBE (lead-bismuth eutectic), lead, etc. A lot of guesses but nothing known precisely. In the case of sodium-to-supercritical CO_2 heat exchangers, envisaged to improve overall plant energy conversion, the consequences of failure of the sodium CO_2 heat exchanger boundary would involve the blow down and intermixing of high-pressure CO_2 in a sodium pool, causing pressurisation which may threaten the structural integrity of the heat exchanger. Available data seem to indicate that the chemical reaction between sodium and CO_2 would likely produce sodium oxides, sodium carbonate (which may cause some clogging of the primary circuit), carbon and carbon monoxide. Information on the kinetics of the sodium–CO_2 reaction is virtually non-existent.	**H, L**

3.2.6 Structural integrity

Integrity of structures and components needs to be ensured during the entire operating time of the reactors. Detection and mitigation of ageing effects that may result from cracking, corrosion, fatigue, embrittlement, etc., is important in view of accident prevention and damage repair capability. In this area, the repair issue was not considered as it refers to an industrial concern.

For SFRs, the main safety issue that calls for improvement is the capability of in-service inspection. Indeed, due to the presence of sodium in the circuits and because of its opacity, in-service inspection of components is quite difficult, especially for the reactor internal core support structures.

This difficulty was highlighted by the French Phénix reactor in the 1990s and innovative remote control devices were developed as a prerequisite for the restart after an emergency shutdown due to negative reactivity events. Developments were also performed for Superphénix for remote control of the reactor vessel and steam generators tubes.

The capability of in-service inspection needs to be improved for both pool and loop SFR concepts (based on MONJU experience).

Two other safety issues were discussed in this area:

- the prevention of sodium leakage in circuits as a potential initiator of sodium fires;
- the rupture of the steam generator envelope as a consequence of tube rupture inside the SGU and sodium-water interaction.

However, as the topics also deal with sodium risks, they were developed and ranked under area E on sodium risks (E5, E6).

So, for this area, the in-service inspection issue was identified with a high safety relevance and with a figure of merit being the detection of loss of structural integrity.

Safety issue **Structural integrity**	**Description/rationale**	**Importance, knowledge levels** **(H, M, L)**
F1 In-service inspection	Inspectability of SFR structures is an important field that calls for improvement (mainly in case of pool-type reactors but also for loop-type concepts). Indeed, due to sodium presence in the circuits and because of its opacity, in-service inspection of components is quite hard (especially for the reactor internal core support structures). Capability of ultra-sonic (US) transmission inside sodium is investigated; US transducers operating at high temperature should be developed and qualified. Development of innovative devices is highly needed for the future concepts. The knowledge level of the issue was ranked M.	**H, M**
F2 Leakage of sodium	Sodium leakage detection should prevent sodium fires (link with E5 topic).	**H, M**
F3 Steam generator tubes	Generator tube rupture may promote failure of the steam generator envelope (as a consequence of sodium-water interaction) with potential hydrogen risk (link with E6 topic).	**H, H(1) and L(2)**

(1) Austenitic steel.
(2) Ferritic steel and other steel.

3.2.7 Other issues

This area covers general safety issues related to reactor innovative design options and new anticipated development of passive and control systems.

For new SFR concepts, with regard to the objective of enhanced safety performance, high importance is given to the reliability and efficiency of passive safety systems and to advanced control instrumentation that aims at preventing accidents or mitigating their consequences.

A first passive system already discussed in the thermo-fluids area (see A3) concerns the generalised natural convection capability that should ensure decay heat removal and avoid generalised core melting (in case of loss of electrical power) or limit consequences of such an accident over the long term.

Other passive systems act against reactivity insertion and propagation of incidental/accident sequences.

In case of severe accidents such as generalised core melting due to a ULOF, formation of large pools of molten materials (fuel, steel cladding) may occur after the primary power excursion, during the transition phase. In those pools, inside the fissile zone, vaporisation of steel may initiate molten fuel motion (for instance coherent sloshing pools) resulting in a re-criticality event and secondary power excursion with high risk of mechanical energy release and potential damage of the vessel structures.

In case of a local melting accident (i.e., sub-assembly accident), propagation of the melting towards the neighbouring sub-assemblies with additional molten fuel is also a concern as, depending on the core characteristics, a re-criticality event may occur and initiate a power excursion.

New studies of passive systems aim at a mastered evolution of the material relocation (i.e., fuel escape outside the fissile zone through specific channels within sub-assemblies, fuel discharge after wrapper failure into specific assemblies or control rod guide tubes, etc.). However, the detailed scenario of events occurring during such a relocation process (conditions of escape, fuel freezing, molten fuel coolant interaction leading to pressure peaks and/or dispersion, etc.) needs to be determined. Moreover, efficiency of the foreseen relocation paths to prevent re-criticality scenarios has to be ensured.

Other systems for in-vessel retention of the core materials (fuel) and long-term coolability are also anticipated in view of reduction of the accident consequences. The core catcher design should ensure corium catching and establishment of a stable material configuration with regard to sub-criticality of the fuel inventory and long term cooling capability. Reliability and efficiency of such systems have also to be ensured.

Another issue concerns the early in-core detection systems in the framework of prevention measures against incident/accident scenarios. For instance, systems allowing early power shutdown in case of detection of an abnormal sodium temperature increase inside the core, should prevent core degradation; delayed neutron detection with a short time response could limit the propagation of a sub-assembly blockage accident beyond the neighbouring sub-assemblies (lessons learnt from the past R&D SCARABEE experimental programme).

The last issue is related to the possible use of alternative fluid to sodium in the secondary circuit and concerns the need to ensure the absence of the energetic interaction with water that could result from steam generator tube rupture.

Within this area, the issues of high importance and their figures of merit (FoM) are the following:

- reliability and efficiency of passive safety systems [FoM: prevention, mitigation provisions against re-criticality events (mechanical energy release)];
- early in-core detection systems (FoM: prevention of accidents);
- interaction of water with secondary fluids different from sodium (FoM: prevention of energetic event in secondary circuit).

The three issues were ranked M for status of knowledge as they refer to innovative systems (design dependent) and/or options that need additional data.

Safety issue Other issues	Description/rationale	Importance, knowledge levels (H, M, L)
G1 Reliability and efficiency of passive safety systems	Adoption of passive systems in the SFR concepts contributes to enhanced safety performance. Passive systems may be designed against reactivity insertion and generalised natural convection capability may improve decay heat removal. In case of severe accidents, passive systems aiming at decreasing the risk of re-criticality by a mastered evolution of material relocation are considered (i.e., fuel escape outside the fissile zone through specific channels inside sub-assemblies, fuel discharge after wrapper failure into specific assemblies or control rod guide tubes…); systems for fuel in-vessel retention and long-term coolability are also studied in view of reduction of the accident consequences (core catcher ensuring corium catching, cooling and sub-criticality of the fuel inventory). Reliability and efficiency of passive systems have to be ensured (design dependent).	**H, M**
G2 Early in-core detection systems	Within the frame of prevention measures, early in-core detection systems should allow early power shutdown and prevention from core degradation (for instance, detection of sodium temperature increase at outlet of sub-assemblies, etc.). Improvement of the delayed neutron detection system aiming at a low response time would be beneficial in case of a sub-assembly blockage accident (avoiding propagation of the accident beyond the neighbouring sub-assemblies). Development of such systems (design dependent) and evaluation of their reliability contribute to improved safety performances.	**H, M**
G3 Interaction of water with secondary fluids different from sodium	In case alternative fluid to sodium is used in secondary circuit, it is important to ensure that no energetic interaction with water can happen as a result of generators tube rupture. No information is now presently available on the characteristics of anticipated fluid choice. Design dependent issue.	**H, M**

3.3 Facilities versus issues

The following table provides a short description of the facilities that have been proposed and ranked by the group members in relation to each safety issue. More detailed information on each facility is given in the specific facility sheets gathered in Appendix 1. Some other facilities are also described in Appendix 1 but not included in the table below as they were not retained nor ranked by the group members.

In order to extend the basis of experimental facilities addressing a maximum number of safety issues the TAREF members also agreed to add a chapter dedicated to relevant Russian experimental facilities provided that appropriate information will be made available in a realistic time frame. Unfortunately, no information could be obtained from the Russian side.

Instead, the Indira Gandhi Centre of Atomic Research (IGCAR), India, kindly provided valuable information on its experimental facilities which may be relevant to the safety issues identified by the TAREF members. However, the information on the Indian facilities was made available to the group late in the project (just before the last meeting of 2-3 February 2010) and no in-depth and direct discussion with the Indian representatives could occur in a way similar to that with the other members. Therefore, in spite of exchanges of e-mails of questions/answers, the Indian experimental facilities are addressed in a separate table (Table 4).

Table 3: Experimental facilities versus safety issues

A. Thermo-fluids			
Issue	**Facility (institution)**	**Availability**	**Capabilities**
A1 Flow regime transitions, transport properties, channel flow distribution, sodium boiling	ALINA (KIT-FZK)	In operation.	Thermalhydraulics of free surface flows, small scale generic heat transfer tests and MHD pump.
	KASOLA (KIT-FZK)	In operation mid-2011.	Pool experiment 1:5 plus bundle flow, heat transfer correlations, new bundle concept.
	Liquid sodium facility NATAN (FZD)	In operation, available shortly.	100 litres of Na; temperature up to 300°C; velocities up to 2 m/s in pipes of 50 mm diameter.
	NADYNE (CEA)	Under design, expected operation in 2017.	TH characterisation scale 1 fuel assembly; possibility to realise fast thermal transients.
A2 Coolant-structure interaction	TTS (JAEA)	In operation.	Thermal transient test facility for structures: thermal creep fatigue, thermal striping.
	KASOLA (KIT-FZK)	In operation mid-2011.	See A1 above.
A3 Natural convection	PDRC (KAERI)	Under design phase, installed end of 2012.	Sodium test loop including the primary heat transport system and the passive decay heat removal circuit scaled down from the KALIMER 600 reactor. Investigation of natural circulation cool-down capability after reactor shutdown and primary pump trips.
	Large Scale Sodium Loop Complex - ATENA (JAEA)	Under construction, 1st operation in 2012, demonstration tests planned in 2014.	Several tests loops for development and demonstration tests of components (60 MW, 20-100 m^3 Na, 250-550°C); capability of simulation of the overall cooling system in JSFR.
	PLANDTL (JAEA)	In operation.	Plant dynamic testing loop for development of SFR components. Thermalhydraulic simulation of primary/secondary loops (flow rates of 1 200/600 1/mn) and reactor vessel (6 bundles of 37 pins connected to upper sodium plenum, maximum heat flux of 2 MW/m^2); temperature 650°C, 10 tons of sodium.
	CYBL (SNL)	Now engaged in natural circulation studies for LWRs; can be converted to sodium.	Cylindrical boiling facility for natural circulation simulation of a full-scale nuclear reactor; 5 MW electrical heating
	KASOLA (KIT-FZK)	In operation mid-2011.	See A1 above.

Table 3: Experimental facilities versus safety issues *(cont.)*

B. Fuel safety			
Issue	**Facility (institution)**	**Availability**	**Capabilities**
B1 Fuel pin performance under steady-state conditions (irradiation)	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	Experimental fast reactor for irradiation tests supporting development of SFR fuels and materials. Irradiation of fuels and materials under representative conditions: maximum neutron flux total/fast (E>0.1MeV) of $5.7*10^{15}/4.0*10^{15}$ n/cm^2/s, Na cooling temperature 350/500 °C (inlet/outlet).
	JHR (CEA)	2017.	New MTR reactor for fuels and materials developments. Irradiation of fuels and materials under significant fast flux: $1.0*10^{15}$ n/cm^2/s for E>0.1MeV; NaK device foreseen, max. temperature 600°C.
B2 Margin to fuel melting	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	Experimental fast reactor for irradiation tests supporting development of SFR fuels and materials. Irradiation of fuels and materials under representative conditions (see B1); slow power transients capability.
	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	Experimental reactor (water pool type) dedicated to fuel safety experiments and extensively used (1976-2001) for SFR studies (international R&D programmes, LOF, TOP transients). Slow power transients starting from nominal reactor conditions (400-600 W/cm) under representative cooling conditions; low uncertainty on energy deposit (± 5%); use of irradiated fuel.
	JHR (CEA)	2017	See B1 above. Power transients up to 600 W/cm/sec with 1% U5.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	Transient reactor test facility (TREAT, graphite reactor) designed to test the behaviour of various fuels and structural materials under accident transient conditions of SFRs (LOF, TOP) starting from zero power. Possible slow power transients under special conditions.
	ACRR (SNL)	Operational.	Annular core research reactor (water pool type) mainly used for radiation effects studies. Potential use for fuel transient testing under abnormal and/or accident conditions of SFRs (slow power transients under specific conditions).

Table 3: Experimental facilities versus safety issues *(cont.)*

Issue	Facility (institution)	Availability	Capabilities
B3 Margin to deterministic pin failure (under slow power transients)	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	See B2 above.
	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	See B2 above.
	JHR (CEA)	2020.	See B1 above; this test is foreseen in a safety dedicated sodium loop, max. temperature 600°C, to be operated in reflector, with displacement systems in a thermal neutron flux up to 5.5 10^{14}n/cm².s
	TREAT (INL)	Shut down; decision for restart expected in 2010.	See B2 above.
	ACRR (SNL)	Operational.	See B2 above.
B4 New fuel pin designs and materials (fuel, cladding)	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	See B1 above.
	JHR (CEA)	2017.	See B1 above, power transient could be studied too with displacement systems.
	ACRR (SNL)	Operational.	See B2 above.
B5 Use of minor actinides (MA)	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	See B1 above.
	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	See B2 above.
	JHR (CEA)	2015.	See B1 above.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	See B2 above.
	ACRR (SNL)	Operational.	See B2 above.

Table 3: Experimental facilities versus safety issues *(cont.)*

C. Reactor physics			
Issue	**Facility (Institution)**	**Availability**	**Capabilities**
C1 Doppler, fuel expansion reactivity feedback	MASURCA (CEA)	2017.	The critical facility MASURCA is dedicated to the neutron studies of fast reactors lattices. The adaptability of the MASURCA core allows the validation of innovative SFR core designs: • Physics and neutron parameters of new lattices (low Pu enrichment, dense fuels, compact reflectors and shielding). • Physics and neutron parameters of large fast reactor cores: zone decoupling, power map control, absorber reactivity worth amplification, instrumentation issues.
C2 Reactivity feedback due to sodium voiding	MASURCA (CEA)	2017.	See C1 above. MASURCA capacities for study: • Void sodium effect reducing (Na upper plenum, moderator introducing). • Degraded or incidental configurations in which several material zones would be redistributed.
D. Severe accidents			
Issue	**Facility (Institution)**	**Availability**	**Capabilities**
D1 Unprotected loss of flow with subsequent transient overpower accident	MELT (JAEA)	In operation.	Out of pile facility for study of molten material behaviour under CDA of SFRs (structure erosion by jets). Use of induction heated crucible with simulating materials (melt of alumina, steel and tin, 2 300°C, 20 l maximum).
	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	Experimental reactor (water pool type) dedicated to fuel accident experiments extensively used (1976-2001) for SFR studies (international R&D programmes). Simulation of ULOFs and TOPs (unprotected loss of flow and transient overpowers) starting from nominal reactor conditions (400-600 w/cm) under representative cooling conditions; low uncertainty on energy deposit (± 5%); use of irradiated fuel.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	Transient reactor test facility (TREAT, graphite reactor) designed to test the behaviour of various fuels and structural materials under accident transient conditions of SFRs (ULOF, TOP) starting from zero power; use of irradiated fuel.
	ACRR (SNL)	Operational.	Annular core research reactor (water pool-type) mainly used for radiation effects studies. Potential use for fuel transient testing under abnormal and/or accident conditions of SFRs (fast power transient from low initial power).

Table 3: Experimental facilities versus safety issues *(cont.)*

Issue	Facility (Institution)	Availability	Capabilities
	IGR (Kazakhstan)	Available.	Impulse graphite reactor (same as TREAT) mainly used for transient tests (non-controlled pulse or controlled pulse mode) using fresh fuel section. UO_2 of about 8 kgs can be molten in the test capsule. Simulation of molten materials behaviour in CDA.
	CAFE (ANL)	Operational.	Core alloy flow and erosion facility. Used for study of materials flow, freezing of metallic-fuel bearing melts, effects of eutectic liquefaction of channel structure (one-dimensional horizontal channel).
D2 Local blockage accident	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	See D1 above.
	IGR (Kazakhstan)	Available.	See D1 above.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	See D1 above.
D3 Slow power transients	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	See B2 above.
	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	See B2 above.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	See B2 above.
	ACRR (SNL)	Operational.	See B2 above.
D4 Fuel coolant interaction	MELT (JAEA)	In operation.	Out-of-pile facility for study of molten material behaviour under CDA of SFRs; also used for FCI in water system. Small sodium loop available. Use of induction heated crucible with simulating materials (melt of alumina, steel and tin, 2 300°C, 20 l maximum).
	PLINIUS-KROTOS (CEA)	Refurbishment needed for sodium use (2013-2015).	Corium-water interaction facility used for LWRs severe accidents study. Molten corium (5kg) dropped into water, triggering steam explosion; need of conversion to sodium under study.

Table 3: Experimental facilities versus safety issues *(cont.)*

Issue	Facility (institution)	Availability	Capabilities
	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	See D1 above.
	IGR (Kazakhstan)	Available.	See D1 above.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	See D1 above.
	ACRR (SNL)	Operational.	See D1 above.
	MCCI (ANL)	Operational.	Large reactor test cell (1 000 m^3) for study of LWR severe accidents issues (radioactive core melt, steam explosion, hydrogen production issues); may be used for fuel-coolant interaction and PAHR issues of SFR.
	SURTSEY (SNL)	Operational.	SURTSEY facility used for LWR severe accident studies. Possible use for large tests with molten materials and their interaction. Large sealed pressure vessel, 1/10th scale; pressure and temperature measurements.
D5 Post-accident decay heat removal	PLINIUS-KROTOS- (CEA)	Refurbishment needed for sodium use (2013-2015).	Corium-water interaction facility used for LWRs severe accidents study. Molten corium (5kg) dropped into water, triggering steam explosion. Potential capability for study of debris formation + need of conversion to sodium: under study.
	MELT (JAEA)	In operation.	See D4 above.
	IGR (Kazakhstan)	Available.	See D1 above.
	PLINIUS-VULCANO- (CEA)	Refurbishment needed for sodium use (2013-2015).	Corium melting facility used for LWR severe accidents study. Molten corium (50-100 kg) capability with molten metals. Potential use for core-catcher design studies + need of conversion to sodium: under study.
	MCCI (ANL)	Operational.	See D4 above.
	ACRR (SNL)	Operational.	See D1 above.

Table 3: Experimental facilities versus safety issues *(cont.)*

Issue	Facility (institution)	Availability	Capabilities
D6 Release of fission products (source term uncertainty)	CABRI (IRSN, operated by CEA)	Restarting operation in 2011 for LWRs. Need of modifications for SFRs tests.	See D1 above.
	TREAT (INL)	Shut down; decision for restart expected in 2010.	See D1 above.
	ACRR (SNL)	Available; operational.	Annular core research reactor (water pool type) mainly used for radiation effects studies. Potential use for fuel transient testing under abnormal and/or accident conditions of SFRs (fast power transient from low initial power).
	VERDON (CEA)	To be operational for SFR conditions in 2014.	Out-of-pile facility for studying irradiated fuel behaviour and fission products release under simulated heat transient. Use of induction furnace under various atmospheres (He, H_2, steam, air) up to 2 700°C. On-line FP measurements.
	MERARG (CEA)	Operational.	Out-of-pile facility for studying irradiated fuel behaviour and fission gas release under simulated heat transient. Use of induction furnace under neutral atmosphere up to 2 700°C. On-line FG measurements.
E. Sodium risks			
Issue	**Facility (institution)**	**Availability**	**Capabilities**
E1 Spray sodium fires	SAPFIRE (JAEA)	Available.	Test platform dedicated to the study of sodium leak accident consequences such as: spray, columnar, pool type fires, sodium concrete interaction, sodium structure interaction with chemical reaction and aerosols behaviour. Test section: 100 m^3, 0.2 MPa, ventilation rate of 70 Nm^3/mn, 10 tons of sodium.
	SURTSEY (SNL)	Operational.	SURTSEY facility used for LWR severe accident studies. Possible use for large tests with molten materials and their interaction, sodium fires and sodium water interaction. Large sealed pressure vessel, 1/10th scale; pressure and temperature measurements, aerosol characterisation.
E2 Pool sodium fires			*Issue not retained.*
E3 Sodium fire aerosols behaviour	SAPFIRE (JAEA)	Available.	See E1 above.
	SURTSEY (SNL)	Operational.	See E1 above.

Table 3: Experimental facilities versus safety issues *(cont.)*

Issue	Facility (institution)	Availability	Capabilities
E4 Sodium-concrete reaction	SAPFIRE (JAEA)	Available.	See E1 above.
	SURTSEY (SNL)	Operational.	See E1 above.
E5 Sodium leak evolution and detection	TTC (SNL)	Available.	Thermal test complex. Possible use for sodium combustion testing.
E6 Sodium-water interaction in steam generators units (SGU)	SWAT-1R (JAEA)	Available.	Facility dedicated to the study of water-sodium interaction due to pipe rupture in a steam generator and to self-wastage behaviour of double tube steam generator pipe. Tank volume: 630 l, Tmax = 580°C, Pmax = 1.96 MPa.
	SWAT-3R (JAEA)	Available.	Facility dedicated to the investigation of the propagation of steam generator tube rupture in prototypical conditions (large scale facility). 15 tons of sodium, T = 555°C, Pmax = 1.96 MPa, tank volume: 10 m^3; water heaters of 3.1 and 4.8 m^3.
	DIADEMO (CEA)	To be operational in 2011.	Development of SWR with hydrogen detection.
E7 Sodium-water interaction in air	SURTSEY (SNL)	Operational.	See E1 above.
E8 Sodium interaction with alternative coolant species	DISCO-2 (CEA)	Available.	Study of sodium-CO_2 interaction, jet of CO_2 into a pot of 2 litres of liquid sodium. Possibility to follow the exothermicity of the reaction by thermocouples plus an on-line spectrometer.
	SURTSEY (SNL)	Operational.	See E1 above.
	STACTF (ANL)	Under process of being made operational.	Sodium technology and advanced component test facility. Dedicated to testing of SFR components and technologies including sodium compatibility of advanced materials (i.e., Na-CO_2 interaction). Test scale: 3-5 m^3.
	LIFUS5 (ENEA)	Refurbishment needed for sodium use.	LIFUS5 loop, designed for study of interaction between heavy liquid metal (lead, LBE) and water (see G3 below), can be modified to study also the interaction with sodium.

Table 3: Experimental facilities versus safety issues *(cont.)*

F. Structural integrity			
Issue	**Facility (institution)**	**Availability**	**Capabilities**
F1 In-service inspection	DOLMEN (CEA)	Available.	Qualification of ISI techniques and instrumentation.
	USV (ANL)	Available.	Under sodium viewing test facility. Evaluations of ultrasonic transducer and imaging system (use of 2 sodium tanks).
F2 Leakage of sodium *link with E5*	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	See B1 above. Prevention, qualification of design. Potential to perform the verification tests of ISI devices.
F3 Steam generator tubes rupture *link with E6*	SWAT-1R (JAEA)	Available.	See E6 above.
	SWAT-3R (JAEA)	Available.	See E6 above.
	SURTSEY (SNL)	Operational.	See E1 above.
G. Other issues			
Issue	**Facility (institution)**	**Availability**	**Capabilities**
G1 Reliability, efficiency of passive safety systems	PLINIUS-VULCANO- (CEA)	Refurbishment needed for sodium use (2013-2015).	See D5 above.
	IGR (Kazakhstan)	Available.	See D1 above.
G2 Early in-core detection systems	JOYO (JAEA)	Stopped in 2007 due to internal damage; repairing plan underway.	See B1 above.
G3 Interaction of water with secondary fluids different from sodium	LIFUS5 (ENEA)	Operating for fusion tests; can be used for SFR in remaining time windows.	LIFUS5 loop for study of interaction between heavy liquid metal (lead, LBE) and water as a result of steam generator tube rupture of fast reactors. Injection of water at the bottom of a tube bundle (closed SS tubes immersed in liquid metal) inside a vessel of 0.1 m^3. Water mass injection: 1-6 Kg in 5-8s, Pmax: 15.5 MPa, T: 265-325°C. Possible study of pressure waves propagation, tube rupture propagation.

Table 4: Indian experimental facilities*

A. Thermo-fluids			
Issue	**Facility**	**Availability**	**Capabilities**
A3 Natural convection	SADHANA (SAfety Grade Decay Heat loop RemovAl in NAtrium)	Operational.	355 kW capacity sodium facility; maximum operating temperature 600°C, maximum flow $5m^3/h$ at 0.5 MPa. Heat removal capacity of DHX and AHX at different sodium temperatures and sodium levels, measurement of air flow distribution in AHX shell side, behaviour of SGDHR during scram and behaviour of SGDHR during station blackout are planned in this facility.

B. Fuel safety			
Issue	**Facility**	**Availability**	**Capabilities**
B1 Fuel pin performance under steady-state conditions (irradiation)	FBTR (fast breeder test reactor)	Available.	Experimental fast reactor for irradiation tests. The current major mission of FBTR is to test irradiate MOX fuel of power reactor composition (29%PuO_2) to a target burn-up of 100 GWd/t at a LHR of 450 W/cm. Irradiation of fuels and materials under representative conditions: maximum neutron flux of $3.15*10^{15}$, sample temperature: 380°C to 490°C.
B4 New fuel pin designs and materials (fuel, cladding)	FBTR	Available.	See B1 above (for irradiation only).
B5 Use of minor actinides (MA)	FBTR	Available.	See B1 above (for irradiation).

* All the mentioned Indian experimental facilities are located at IGCAR (Indira Gandhi Centre of Atomic Research).

Table 4: Indian experimental facilities *(cont.)*

D. Severe accidents			
Issue	**Facility**	**Availability**	**Capabilities**
D4 Fuel coolant interaction	SOFI	Available.	A facility has been built for the experimental simulation of molten fuel coolant interactions after a core disruptive accident which involves melting of the grid plate by the molten fuel, heat transfer between dislocated molten fuel with surrounding sodium, solidification of molten fuel followed by fragmentation, settlement behaviour of core debris on core catcher and heat transfer from debris surface to the sodium pool. These simulations will be carried out in various phases. In the first phase, simulants like woods, metal will be used in a water system. In the second phase, molten 304 L and sodium are used and in the third phase molten UO_2 and sodium will be used. For simulating decay heat, heating coils will be provided on the surface of the core catcher. Imaging of core debris dispersion patterns on the core catcher in case of sodium experiments is being developed. In the test during phases 2 and 3, core debris movement will be visualised.
D5 Post-accident decay heat removal	SOFI	Available.	See D4.
E. Sodium risks			
Issue	**Facility**	**Availability**	**Capabilities**
E1 Spray sodium fires	MINA	Available.	Simulation of a sodium fire (pool as well as spray fire), involving a few grams of sodium (spray) and a few tens of kilograms of sodium (pool) to understand the basic science of sodium fires. The facility consists of a rectangular steel chamber (6 m x 6 m x 5 m) which has integrated sodium loops for injecting the required amount of sodium. It is provided with state of art instrumentation for measuring the relevant parameters.
	SFTF (sodium fire test facility)	Available.	The second facility is for carrying out the sodium fire studies on a bigger scale involving quantities up to 500 kg (mixed spray and pool fire). This facility will be used for qualifying sodium leak collection trays, sodium fire fighting system, sodium concrete interaction and sodium aerosol behaviour in a larger volume.

Table 4: Indian experimental facilities *(cont.)*

Issue	Facility	Availability	Capabilities
E3 Sodium fire aerosols behaviour	SFTF (sodium fire test facility)	Available.	See E1.
E4 Sodium-concrete reaction	SFTF (sodium fire test facility)	Available.	See E3.
E6 Sodium-water interaction in steam generators units (SGU)	SOWART (sodium water reaction test facility)	Available.	This facility was constructed to study the behaviour of self wastage and leak enlargement during sodium-water reaction, to study impingement wastage and to develop leak detection methods. Self wastage studies were carried out at different steam leak rate in the range of 10-50 mg/s. Model cold trap testing was completed in this test facility. Different hydrogen meters were calibrated by injecting known quantities of hydrogen. Studies on a large number of SG tube material specimens for adjacent tube wastage studies, self wastage studies and development and testing of in-sodium and cover gas hydrogen meters are planned in this facility.
F. Structural integrity			
Issue	**Facility**	**Availability**	**Capabilities**
F1 In-service inspection	LCTR (large component test rig)	Operational.	The large component test rig (LCTR) was constructed to carry out full scale testing of a few critical components of the prototype fast breeder reactor (PFBR) in sodium under simulated reactor operating conditions.
F2 Leakage of sodium *link with E5*	LEENA	Operational.	This facility was constructed for the performance evaluation of the wire type leak detector layout of the PFBR (prototype fast breeder reactor) secondary circuit by creating sodium leak. Different experiments were conducted in different test sections at 350 and 550°C. Lowest leak rate that could be simulated is 214 g/h.
G. Other issues			
Issue	**Facility**	**Availability**	**Capabilities**
G2 Early in-core detection systems	LCTR	Operational.	See F1 above.

4. SUMMARY AND RECOMMENDATIONS

4.1 Summary

The Task Group on Advanced Reactor Experimental Facilities (TAREF) was initiated based on discussions held by the NEA Committee on the Safety of Nuclear Installations (CSNI) and the NEA Committee on Nuclear Regulatory Activities (CNRA) during a joint workshop on the Role of Research in a Regulatory Context (RRRC-2, June 2007). Among other topics, the workshop addressed the challenges that the nuclear community will face when performing safety evaluations of advanced reactor designs, the research that may be needed to perform the reviews, and the possible means for jointly conducting this research. In particular, the workshop participants discussed research topics relevant to GCRs and SFRs and recommended that the CSNI organise a task group to identify the needed research and recommend a path forward.

The CSNI initiated TAREF to provide an overview of facilities suitable for carrying out the safety research that was considered necessary for GCRs and SFRs. Other reactor systems could be considered in a subsequent phase.

The TAREF was created in spring 2008 with the following participating countries:

- Canada
- China
- Czech Republic
- Finland
- France
- Germany
- Hungary
- Italy
- Japan
- Republic of Korea
- United States

India provided useful information on its facilities for the last meeting of the group (in February 2010) and this information was considered for the conclusions of the task.

The group decided to build on the experience of a similar activity conducted by the CSNI and described in the report entitled *Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SFEAR)*, which focused on facilities suitable for current and advanced water reactor systems. In particular, the SFEAR method was adopted, consisting of first identifying high-priority safety issues that require research, and then categorising the available facilities in terms of their ability to address the safety issues.

At the first TAREF meeting, it was stated that the SFR-related task would need several discussions for elaboration of a PIRT-like approach (PIRT exercise = Phenomena Identification and Ranking Tables) unlike the GCR-related task that could benefit from the already available PIRT compiled in an earlier US-NRC exercise. Hence, the GCR report was issued, first in 2009.

As an initial step, the TAREF participants compiled a questionnaire regarding the main technical issues relevant to SFR safety and deserving further R&D work, and the technical infrastructure that is potentially suitable for experimental studies on SFR safety.

Based on the participants' answers to the questionnaire and consistent with the SFEAR report, the Task Group identified the following technical areas to be addressed:

A. thermo-fluids;
B. fuel safety;
C. reactor physics;
D. severe accidents;
E. sodium risks;
F. structural integrity;
G. other issues.

The technical areas A, B, C and D address phenomena and issues that are specific to the nuclear industry. The other three technical areas E, F, and G address phenomena that are relevant to the nuclear industry, but for which experience may be broader than nuclear.

Other technical areas such as seismic assessment (except for potential consequences on core compaction), instrumentation and control, and human and organisational factors are not treated here, since they are not specific to the nuclear industry and – within the nuclear area – not specific to SFRs. They were addressed only in broad terms in the SFEAR report, which can be consulted for generic information regarding these areas. Other technical areas such as fuel fabrication, fuel handling and irradiated material investigation techniques (as used in hot cell facilities) were not considered as they are more related to operational concerns or not specific to SFRs (hot cells).

For each of the above technical areas (A to G), the task members agreed on a set of safety issues needing research and established a ranking with regard to safety relevance (high, medium, low) and status of knowledge based on the following scale relative to full knowledge: high (100%-75%), medium (75%-25%), low (25%-0%). Only the issues identified as being of high safety relevance and for which the state of knowledge is low or medium were included in the discussion, as these issues would likely warrant further study.

For each of the safety issues, the TAREF members identified the related facilities that were deemed appropriate to address the issue in question, providing relevant information such as operating conditions (in- or out-of-reactor), operating range, description of the test section, type of testing, instrumentation, current status and availability, uniqueness, etc.

Based on the information that was assembled on both safety issues and related facilities, the task members assessed prospects and priorities for SFR safety research and developed recommendations as to priorities and options for the CSNI regarding facility utilisation through programmes that could be pursued internationally. In particular, the group agreed on the main criteria for priority setting, which were based on the following items [high, medium or low (H, M, L) for each item]:

- relevance of the facility to cover a specific issue;
- uniqueness (e.g., one of a kind for in-pile testing);
- availability for a potential programme addressing the issue; three time windows were considered: 0-3 years, 4-8 years, more than 8 years;
- readiness (e.g., staff available to run it);
- operating cost (<0.3; 0.3-1; >1 million USD), or construction cost (<0.5; 0.5-2; >2 million USD);

The group rated those facilities that were costly to either operate or construct as being ranked high in this category as they were more suitable to host a multilateral co-operative programme than facilities of lower cost that could be supported by one country without the need to organise a collaborative programme.

TAREF members who had proposed facilities were requested to characterise their proposed facilities in relation to the above criteria. Based on this, the group's recommendations for the CSNI were developed.

4.2 Conclusions and recommendations

1. The TAREF task proved to be a useful exercise for gathering consensus on the technical areas and issues related to the safety of SFR systems, as well as for identifying a number of facilities that are or will become available in OECD member countries for supporting SFR safety research.

2. Due to the specific context of SFR development characterised by a large R&D activity during the period 1970-1995, then followed by a slow-down period and presently restarting in several countries, a large number of facilities operating in the past with sodium as coolant are no longer available (decommissioned, stopped, on standby) or have been converted to water reactor related purposes. This explains why the availability of relevant facilities for all technical areas is limited in the short-term period, and that the decision to restart or to modify some facilities is under consideration for some ones. This context also led the group to rank the facilities over three time windows, up to the long term (beyond 8 years from now).

3. Based on the responses received, the highest ranked facilities were identified. These facilities are listed in the three tables for the short, medium and long term (see Tables 5 to 7); it should be noted, however, that the presently planned availability of the facilities under construction, to be restarted or refurbished may not be very precise.

4. It is assumed that facilities available in the short term can be used in the medium and long term; similarly, facilities available in the medium term are assumed to be potentially usable in the long term.

5. The group members agreed that for new SFR projects, the most important and top-tier R&D safety needs concern the technical areas with the following priority order:

   - Fuel safety (B) and severe accidents (D) issues are of prime interest due to the lack of knowledge on new pin design and materials.
   - Thermo-fluids (A) and reactor physics (C) issues are of second priority as one can live with the current knowledge when considering some margins to cover uncertainties.
   - Sodium risks (E) and structural integrity (F) issues may be considered with third priority as they are more design dependent.

6. The need for fuel pin irradiation capabilities under representative conditions of fast neutron flux has been identified as a crucial point for addressing safety issues of high priority.

7. In the short term:

   - The Indian FBTR fast reactor can be a valuable resource for irradiation of SFR fuel pins and new materials data; the American reactor ACRR (US DOE) would address issues

related to fuel safety and severe accidents under specific conditions (provided there is confirmation of its availability for testing in the short term).

- The German KASOLA (KIT) facility would provide data for the thermo-fluids issues in relation with the CFD modelling approaches.
- The Japanese SWAT-1R -3R facility (JAEA) can be appropriate for studying sodium water interaction in steam generator units; the Indian SFTF facility can be valuable for addressing several issues related to sodium fires; the SURTSEY facility (US-DOE) can be relevant to studies on sodium fires and sodium-water interaction in steam generators.

8. In the medium term:

- The JOYO fast neutron reactor (Japan, JAEA) was identified as suitable to address fuel safety issues related to new fuel pin design (fuel pin performance and new materials database under irradiation, margin to fuel melting, impact of use of minor actinides) and some other issues; however, uncertainty still exists as a decision on the possible repair and operating schedule has not yet been taken.
- Severe accident issues can only be addressed in a comprehensive way for the medium-term period and beyond due to the lack of available facilities for simulation of representative transient conditions in short term with irradiated fuel pins. IGR (Kazakh facility used for JAEA programmes) which is addressing fresh fuel (controlled fuel relocation, debris bed formation) may be a suitable solution in the medium term as plans are under consideration for it to handle irradiated fuel. The VULCANO (France, CEA) can also help for severe accident issues, provided it is refurbished for sodium use. The TREAT experimental reactor (US DOE) was also considered in the medium term for its relevance to severe accident issues (past experimental programmes simulating fast power transients) but the restart of the facility has not yet been decided (decision expected in 2010).
- The MASURCA (France, CEA) may be suitable for core physics issues for providing improved nuclear data of core materials (in relation with high burn-up level, use of minor actinides) and associated uncertainties.

9. In the long term:

- The French ASTRID SFR prototype, although at first designed as an industrial prototype to be transposed to a future first of a kind commercial reactor, will offer some irradiation capabilities and may address mainly fuel performance issues (new cladding materials test and impact of minor actinides under fast flux). The JHR material testing reactor (a new French facility under construction, CEA) may address fuel safety issues (new materials database under irradiation, impact of use of minor actinides, slow transients under specific conditions). The availability for first testing is foreseen in 2017-2020.
- The CABRI experimental reactor (operated by the CEA for the IRSN R&D programmes) was recognised by the group members as the most appropriate facility to address irradiated fuel behaviour under incidental and accidental conditions (fuel safety issues such as margins to fuel melting and deterministic pin failure, severe accident issues such as consequences of various accidents leading to fuel melting, with associated consequences and risk of critical events and energy release); the facility may be available for testing from 2020 (after completion of LWR dedicated safety programmes).
- In the case of innovative design for secondary circuits, the LIFUS5 Italian facility (ENEA) would address sodium interaction with alternative coolant species.

10. Relevant CSNI working groups should be encouraged to share modelling information and discuss modelling activities relevant to SFR safety in order to help focus the potential test programmes and/or enhance the data utilisation for model development.

11. An activity in the field of severe accidents and fission product behaviour in an SFR environment should be considered in the Working Group on Analysis and Management of Accidents (WGAMA), which has advanced reactors on its agenda. This activity may consist of an international standard problem regarding SFR safety issues. This activity could help define medium-term initiatives (3-5 years) for an analytical or experimental international programme in specific areas of interest.

12. In a similar way that is intended for gas-cooled reactor fuel, the Working Group on Fuel Safety (WGFS) should consider a work frame (workshop or technical discussion) to address the safety aspects of SFR advanced fuel designs, including SFR fuel safety needs and WGFS initiatives for addressing high-priority issues through possible use of one of the facilities available in the short- to medium-term periods.

13. The CSNI is to maintain an adequate level of exchange with the CNRA regarding needs and initiatives in the SFR safety area.

Table 5: Highest ranked facilities – short-term availability: 0-3 years

Issue / Country	Thermo - fluids	Fuel safety	Core physics	Severe accidents	Sodium risk	Structural integrity	Other issues
France				MERARG (D6)	DIADEMO (E6) DISCO -2 (E8)	DOLMEN (F1)	
Germany	KASOLA (A1, 2, 3)						
India	SADHANA (A3)	FBTR (B1, 4, 5)			SOWART (E6) SFTF (E1, 3, 4)	LCTR (F1) LEENA (F2)	LCTR (G2)
Japan	TTS (A2) PLANDTL (A2)			MELT (D1, 4, 5)	SAPFIRE (E1, 2, 4) SWAT1 -3R (E6)	SWAT1 -3R (F3)	
Kazakhstan				IGR (D1, 4, 5)			IGR (G1)
United States	CYBL (A3)	ACRR (B2, 3, 4, 5)		ACRR (D1, 3, 4, 5, 6) MCCI (D4, 5) CAFE (D1) SURTSEY (D4)	TTC (E5) SURTSEY (E1, 3, 4, 6, 7, 8)	USV (F1) SURTSEY (F3)	

Table 6: Highest ranked facilities – medium-term availability: 4-8 years

Issue / Country	Thermo - fluids	Fuel safety	Core physics	Severe accidents	Sodium risk	Structural integrity	Other issues
France	NADYN E (A1)		MASURCA (C1, 2)	KROTOS, VULCANO (D4, 5) VERDON (D6)			VULCANO (G1)
Italy							LIFUS5 (G3)
Japan	LSS-LC (A3)	JOYO (B1, 2, 3, 4, 5)		JOYO (D3)		JOYO (F1)	JOYO (G2)
Korea	PDRC (A3)						
United States		TREAT (B2, 3, 5)		TREAT (D1, 2, 3, 4, 6)			

Table 7: Highest ranked facilities – long-term availability > 8 years

Issue / Country	Thermo-fluids	Fuel safety	Core physics	Severe accidents	Sodium risk	Structural integrity	Other issues
France		CABRI (B2, 3, 5) ASTRID (B1, 4, 5) JHR (B1, 2, 4, 5)		CABRI (D1, D2, D3, D4, D6)			
Italy					LIFUS5 (E8)		

List of issues used in Tables 5 to 7

A1: Flow regime transitions, transport properties, channel flow distribution, sodium boiling
A2: Coolant-structure interaction
A3: Natural convection

B1: Fuel pin performance under steady-state conditions (irradiation)
B2: Margin to fuel melting
B3: Margin for deterministic pin failure (under slow power transients)
B4: New fuel pin designs and materials (fuel, cladding)
B5: Use of minor actinides (MA)

C1: Doppler, fuel expansion reactivity feedback
C2: Reactivity feedback due to sodium voiding

D1: Unprotected loss of flow with subsequent transient overpower accident and uncontrolled passage of gas bubble accident
D2: Local blockage accident
D3: Slow power transients (ATWS)
D4: Fuel coolant interaction
D5: Post accident decay heat removal
D6: Release of fission products (source term uncertainty)

E1: Spray sodium fires
E2: Pool sodium fires
E3: Sodium fire aerosols behaviour
E4: Sodium-concrete reaction
E5: Sodium leak evolution and detection
E6: Sodium-water interaction in steam generators (SGU)
E7: Sodium-water interaction in air
E8: Sodium interaction with alternative coolant species

F1: In-service inspection
F2: Leakage of sodium
F3: Steam generator tubes

G1: Reliability and efficiency of passive safety systems
G2: Early in-core detection system
G3: Interaction of water with secondary fluids different from sodium

References

Communication from IGCAR on Indian facilities for SFR.

"Experimental Facilities for Gas-cooled Reactor Safety Studies", NEA/CSNI/R(2009)8.

K. Gibson *et al.* (2009), Communication to TAREF regarding SFR metallic fuel characteristics (November).

Appendix 1

DESCRIPTION OF EXPERIMENTAL FACILITIES FOR SFR SAFETY STUDIES

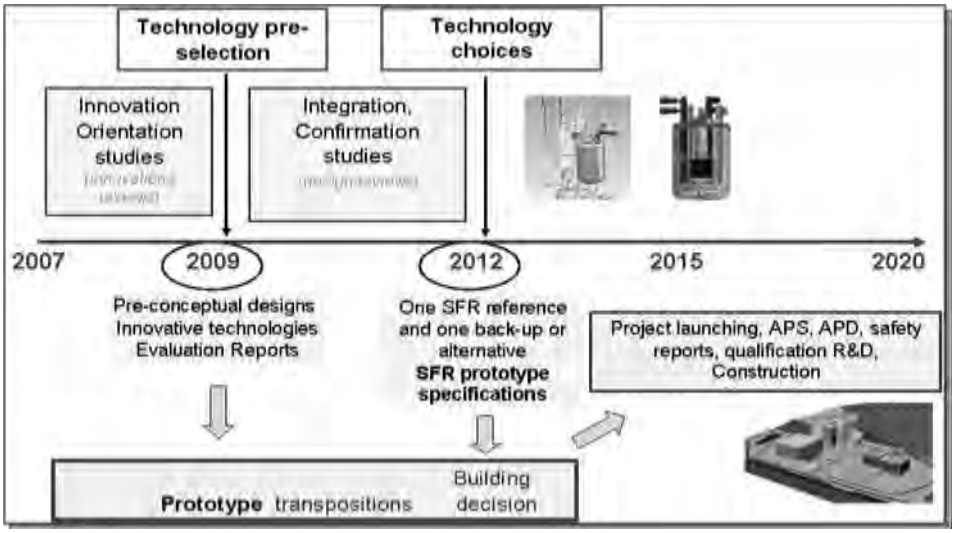
Facilities at the *Commissariat à l'énergie atomique (CEA)*, France

The French SFR R&D facility platform – status in February 2009

The launch of an SFR project in France for a Generation IV prototype reactor is based on a detailed schedule and milestones (see figure below). The SFR French programme is structured within four R&D domains of innovation:

- Project 1: A core with enhanced safety.
- Project 2: Improved resistance to severe accidents.
- Project 3: Energy conversion systems to minimise sodium risks.
- Project 4: Simplified and optimised plant and system design.

SFR French programme: 2009 and 2012 milestones



Regarding these four domains, a list of R&D programme developments have been established in every field (i.e., safety, mechanic, technology, thermo-hydraulic, design qualification). In correspondence, a series of facilities exist at CEA. These facilities were originally devoted to SFR or could be adapted based on specific SFR demand. The objective relies on in making the adequacy between the R&D programme of work related to the four domains, and the facility fleet capability. This work is done in order to define the SFR R&D facility platform.

This task has lead to a list of facilities that have been identified and are considered as necessary to restart, to renovate, or to adapt to the programme demand. In some cases no facility exists or could fit to the demand and some facility design and realisation is therefore under development. In parallel, an evaluation of the international facility fleet and the possibility to use them for R&D purpose is under progress (TAREF is one example of this international approach).

The sodium facility technological fleet is based today on a list of facilities that are necessary to support the SFR programme. Today this evaluation is taking into account a fleet of approximately 20 facilities that can respond to the R&D programmes. Some of the most representative ones are briefly described hereafter.

DOLMEN • Objective: qualification of ISI&R techniques and demonstration • State: in operation. • Related project: No. 4.
DISCO2 • Objective: Na-CO_2 interactions for safety cases. • State: under adaptation to the programme demand. • Related project: No. 2 and 3.
MININANET • Objective: innovative cleaning process of fuel assemblies. • State: under adaptation to the programme demand. • Related project: No. 1 and 4.
TERMINATOR • Objective: sodium technology qualification. • State: under renovation. • Related project: No. 2 and 4.
DIADEMO NA • Objective: qualification of sodium innovative instrumentation and classical instrumentation for calibration. • State: new construction. • Related project: No. 2 and 3.

This work is under permanent evaluation:

- to adapt the facility fleet to the R&D needs and new requirements;
- to adjust the facility purpose to the SFR project progress and choices;
- to pursue a search in synergy with the international programmes.

CEA PLINIUS platform – severe accident experiments with prototypic corium

1. Project title

Facility devoted to test severe accident with core melting.

2. Objective

Severe accident prevention will necessitate the development of alloying means to exclude highly energetic sequences in case of core melting. It will also necessitate proposing robust passive solutions for core catching and evacuation of the residual power in case of a severe accident.

3. Relevance to deploying SFR prototype and potential benefits

A new design of sodium fast reactor core including the potentiality of burning minor actinides and a better approach of passive safety in the case of a severe accident, such as core melting, support a new design of the core catcher associated with innovative ideas.

It could also be used for out of pile tests with prototypic material that would be necessary to study the severe accident behaviour of innovative systems.

4. Infrastructure needs

The CEA has a platform for the testing and simulation of severe accidents. This platform is called PLINIUS, the purpose would be to adapt this facility to sodium core melting tests.

The PLINIUS platform (www.plinius.eu) is today made of:

- VULCANO: VULCANO is a 50-100 kg corium melting facility. It is possible to melt oxides and metals (in depleted uranium) and to mix them for a VULCANO experiment.
- COLIMA: COLIMA is a small scale (few kg) facility with induction heating (up to 170 kW) and a thermostatic 1.5 m^3 enclosure in which it is possible to monitor the gas atmosphere. COLIMA has been up to now used to study aerosols, material interactions, physical properties, etc.
- KROTOS: KROTOS is a corium-water interaction facility in which up to 5 kg of molten corium is dropped into water. Energetic steam explosions can be triggered and studied.
- VITI: VITI is the PLINIUS smallest scale (~10-100 g) facility. VITI is suitable to small scale interaction experiments and to study thermophysical and thermochemical properties. A levitating droplet device has been developed to measure corium viscosity and surface tension. Crucible tests have also been performed in VITI. It will soon be used to test the interaction of potential SFR core-catcher sacrificial materials with UO_2-Fe and later Na.

5. Key dates

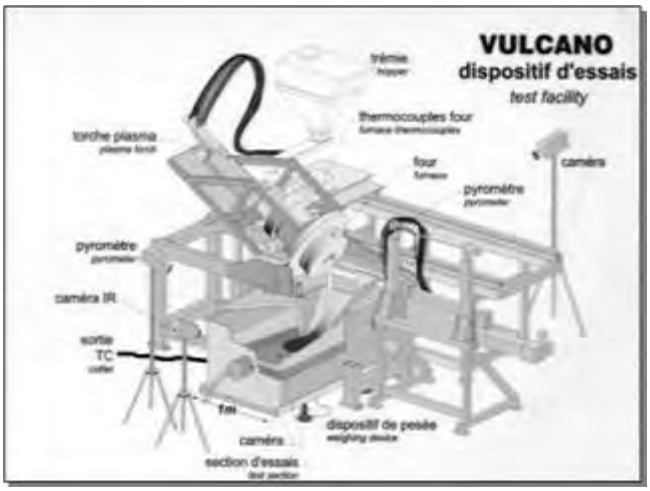
Without sodium: the platform is ready for testings (2009).

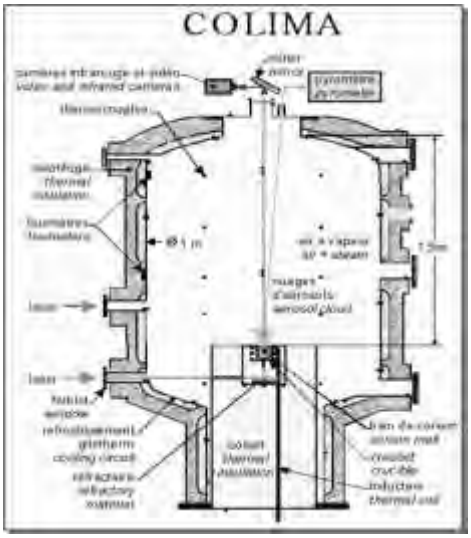
With sodium: it must be defined which facility is the most suitable from sodium testings. The date of modifications is planned for 2013-2015.

6. Location

CEA, Cadarache centre, France

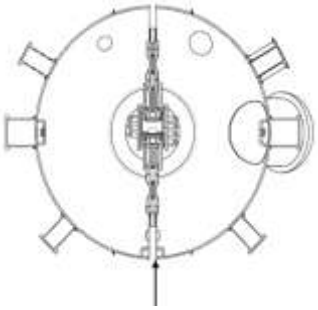
Schematic of the facility





KROTOS

VITI



CEA SET facility: large-scale facility for heat exchange testing

1. Project title

Large-scale facility for heat exchange testing.

2. Objective

To test or demonstrate innovative heat exchangers in the following potential configurations: Na/Na heat exchange, Na/alternative fluid (i.e., non sodium liquid metal or molten salt) heat exchange, steam generators, Na/air heat exchange for DHR components. All of these tests must be done at a significant scale to be able to realise the validation of a reactor component: Therefore, the thermal power is defined in a range from 1 to 10 MWth and the objective is to test a set of heat exchanger tubes at scale 1 (mono-tube and multi-tubes for stability studies).

3. Relevance to deploying SFR prototype and potential benefits

In the frame of heat exchange, all of the previous steam generator designs have been tested and validated on a facility with a significant heat power: 45 MWth for Superphénix SG and one straight tube scale one for EFR project. Development and study of advanced heat exchanger/steam generators are key challenges for future commercial SFRs.

4. Infrastructure needs

The SET facility is planned to be returned to service after a complete check of the loop infrastructure and re-commissioning of the water loop and water treatment as well.

This facility is located in France at CEA/Cadarache.

5. Key dates

Refurbishment: to be defined.

First series of tests: to be defined.

CEA TRIPOT: large-scale multipurpose facility for components demonstration

1. Project title

Large-scale sodium test pot for multipurpose sodium component validation

2. Objective

Grand purpose

To test or demonstrate components, demonstration of devices in sodium condition at scale 1 or at a significant scale for reactor demonstration.

Examples of target equipments

- under sodium fuel handling machine;
- fuel transfer machine;
- in gas fuel transfer cask;
- complementary shutdown systems;
- passive shutdown systems;
- large flow electromagnetic pumps;
- control rods drive mechanism;
- scale 1 ISIR testing.

3. Relevance to deploying SFR prototype and potential benefits

This type of testing has been carried out for specific in-vessel components for all SFRs and shall be essential to test new components, innovative options or enhanced passive safety systems of future advanced sodium fast reactor systems.

4. Infrastructure needs

The TRIPOT facility (three large pots of sodium) is planned to be returned to service after a complete check of the loop infrastructure. In addition to this loop, it must be associated with a large sodium cleaning facility that is able to clean all components after their stay in a sodium pot. This cleaning facility, using a controlled water mist to clean sodium residues under inert gas, also exists and is known as HYPERNANET. This set of facilities is located in France at CEA/Cadarache.

5. Key dates

Refurbishment: 2012.

First series of tests: 2014.

6. Location

CEA, Cadarache centre, France.

TAMARIS

CEA has a platform for seismic testing called TAMARIS and has the capability to validate the GFR designs. The TAMARIS infrastructure and its main shaking table AZALEE belong to the Seismic Laboratory which has 40 years of experience in experimental and analytical earthquake engineering.

The testing facility, TAMARIS, is part of the Seismic Laboratory and has a staff of about 20 researchers and technicians. Its objectives are model development and validation, calculation methods development and qualification, codification, seismic qualification of components, and assessment and retrofit of existing facilities. TAMARIS infrastructure, which was opened in 1988, is characterised by:

- The capacity of the AZALEE shaking table with 100 tons capacity (allowable model mass) is the largest shaking table in Europe. At this time, tests with masses up to 92 tons have been successfully performed. This 6 m x 6 m and 6 degrees of freedom shaking table allows testing of specimens under independent excitations of any kind: sinusoidal, random, shock and time history with 0-100 Hz frequency ranges. Maximum accelerations of 1 g in the horizontal and 2 g in the vertical directions can be applied to specimens approaching the maximum payload of the table.
- Three other smaller shaking tables with similar maximum acceleration, velocity and displacement are available but reduced capacities in term of mass and degrees of freedom are used for qualification and research experimental programmes.
- A high quality control and acquisition system allows recording and processing of 256 channels and is linked with a scientific computing and processing system used for the definition and the accomplishment of tests and subsequent interpretation. In recent years, the infrastructure equipment has been permanently upgraded (a new digital controller was installed for AZALEE during 2008).

The laboratory is part of a service with about 100 engineers, scientists and technicians involved in different fields of mechanical engineering (static, dynamic, vibrations, fluid-structure interaction, fracture mechanics, material engineering, computer science, code development, etc.).

A generic finite element computer code CAST3M is designed, developed and used in support of the R&D activities, especially for preparation and interpretation of the experimental campaigns. This

homemade FEM programme is largely used and developed in universities and R&D centres in France and Europe (e.g., JRC Ispra, University of Porto). Important developments and applications concern nuclear equipment with fluid structure interaction, contact and material nonlinearities. This software also represents an opportunity for paving the way towards hybrid testing whose importance is increasing in experimental earthquake engineering.

TAMARIS is presently available.

CEA-MASURCA critical mock-up facility

1. Project title

A flexible ZPR for neutron physic studies for SFR.

2. Objective

MASURCA is a "zero power" air-cooled critical facility (5 kW) dedicated to the studies of fast neutron reactors. The facility started in 1966, and is now engaged in a refurbishment programme.

The MASURCA core consists of an arrangement of sub-assemblies, having a length of about 3.8 metres, suspended to an upper plate structure that allows building large cores with a diameter greater than 3 metres. The sub-assemblies are made of a stainless steel wrapper containing a mixture of small fissile and non fissile elements, rodlets and platelets of various dimensions that are assembled according to representative lattices, with a very high flexibility.

MASURCA: Bottom view of the experimental core preparation

3. Relevance to deploying SFR prototype and potential benefits

Experiments dedicated to the physics investigation of innovative SFRs:

- Detailed neutronic characterisation of tight-pitch oxide (and/or carbide) fuel pin lattices with a low plutonium fraction, variations of the plutonium isotopic vector, central substitutions of various materials, local spectral variations, sodium core (and plenum) partial/total void effects, streaming effects, peripheral core interface studies, etc.
- Experiment dedicated to innovative reflectors and shields analytic investigations of neutron/gamma propagation and heating in various configurations to various types of fast reactors.

- Experiments dedicated to generic studies of large fast reactor cores, with special emphasis on phenomena typical of loosely-coupled radial cores: power tilts, control rod amplification/interaction, core instrumentation and detection of anomalies (core loading error), etc.
- Experiment dedicated to criticality measurements of simulated degraded core and accidental configurations, in which several material zones would be redistributed.

4. Infrastructure needs

One of MASURCA's main assets is the large diversity of materials available in the facility to simulate different fast reactor cores. In particular, there is at the MASURCA facility an important inventory of uranium and plutonium fuels. This diversity, combined with the sub-assembly loading and instrumentation flexibility (fission and gamma chambers, activation foils) makes it possible to investigate a large variety of lattices (spectra) and configurations. Multiple physics phenomena can thus be studied separately. This feature has been largely used in the past (e.g., neutronic validation for Phénix and Superphénix reactors).

In future experiments, advantage will be taken of an upgraded and modernised facility, as well as of advances in measurement and simulation techniques. Significant progress is expected in the reduction of some experimental uncertainties: X and gamma dosimetry, reactivity measurements, new fission chambers, new electronic and data acquisition system, new processing software, In addition, the possibility of new types of measurements, such as Doppler measurements on small samples in MASURCA, should be studied.

5. Key dates

MASURCA availability after refurbishment: 2017.

6. Location

CEA, Cadarache centre, France.

Jules Horowitz reactor (MTR)

1. Project title

Large innovative MTR reactor.

2. Objective

The JHR is a 100-MWth, tank-type reactor. The design of the JHRs core meets a two-fold objective:

- *It allows a high fast-neutron flux inside the core*, obtained through a high power density within the core (600 kW/l). Consequently, the JHR's core (600 mm high) needs pressurised forced convection water cooling and is held in a pressure vessel (740 mm outer diameter). The experimental devices may be loaded in the central cavity of the fuel elements (which consists of concentric circular plates) or may replace some fuel elements, when more room is needed.
- *It allows a high thermal neutron flux in the beryllium reflector* that encircles the core's pressurised vessel. Six radial channels filled with non-pressurised water are cut into the reflector and may house displacement systems. The experiments may have a fixed position in the reflector, or may use the displacement systems that can achieve, depending on the experimental objectives, a fine tuning of the power or various transients (up to 600 W/cm/sec with 1% U5).

The reactor building accommodates the primary circuit and the experiment area communicating with the core, catering to the running of some 10 experimental loops. Next to the bunkers housing these loops, a fission product analysis laboratory allows the on-line characterisation of the content of the fluids yielded by fuel samples. Communicating with the reactor building, the nuclear ancillary building houses the pools, laboratories, and hot cells for reactor operation, and experimental processes. The latter requires, aside from a cell for feed fuel management, two hot cells, for the preparation and treatment of experimental devices, before and after irradiation. In addition, a so-called alpha cell serves as a dedicated cell for experiments investigating fuel operating limits (i.e., safety experiments that may result in clad failures). The capability for investigation of degraded fuels, possibly right up to melting, whether involving interaction with the coolant or otherwise, is a significant improvement.

3. Relevance to deploying SFR prototype and potential benefits

Development and qualification of SFR fuels in nominal and transient conditions:

- Margin to melt during the fuel cycle.
- Transient behaviour: possible slow power transients under special conditions with advanced instrumentation/cladding deformation under rapid transient.
- Evolution of a pre-rupted cladding under irradiation.
- Development and qualification of MA SFR fuels.
- Development and qualification of carbide fuels and absorbing materials.
- Demonstration of safety passive shutdown systems.
- Development and qualification of SFR instrumentation (acoustics, neutronics, thermics).
- Qualification of cladding and structure materials.

4. Infrastructure needs

To meet the above mentioned SFR needs, two kinds of new experimental devices are envisioned:

- a SFR instrumented capsule, using NaK as coolant (i.e. for margin to melting measurement); and
- a SFR integrated loop, using <u>sodium</u> as coolant, devoted to the safety experiment and designed to handle the high activity that may result from failed cladding experiments.

Both devices will be devoted to single rod (possibly pre-irradiated) studies (e.g., oxides, carbides, or transmutation), and are foreseen to be able to: hold a neutronic shield, use the displacement system and to be located in the core or in the reflector.

5. Key dates

- 2015 – JHR availability for experimental irradiations.
- 2017 – SFR capsule.
- 2020 – SFR safety loop.

6. Location

Under construction in CEA Cadarache centre, France.

CEA – MERARG facility

1. Project title

MERARG is able to reproduce temperature transients on an irradiated LWR fuel pellet. After a complementary safety assessment of the facility to take into account the SFR fuels conditions, it could be used for SFR fuels tests.

2. Objective

To address the safety issue of potential FP releases by:

- studying the fission gases release mechanisms;
- quantifying the fission gas source term in case of accidental sequences (e.g., LOCA for PWR); or
- measuring the fission gas total inventory in the fuel sample.

3. Relevance to deploying SFR prototype and potential benefits

This facility is operational in a dedicated hot cell of the LECA-STAR laboratory at the CEA Cadarache centre. A first version of this facility is already operational for LWR fuels. In the future the MERARG facility could also be used to qualify fuels for GEN-IV reactors.

4. Infrastructure needs

Facility main features

- Maximum temperature: plateau up to 2 800°C.
- Temperature maximum increase rate:
  - controlled temperature increase rate from 0.05°C up to 50°C/s;
  - power injection for accelerated ramps from 50°C/s up to 200°C/s;
- Circulating atmosphere in the furnace:
  - ar, He and air with a Pt crucible;
  - ar or He with a W crucible.
- Fission gas release measured by:
  - on-line gamma spectrometry;
  - on-line micro gas chromatography (stable isotopes);
  - pre-voided capacity and trap for gas storage and delayed measurement by gas chromatography and mass spectrometry.
- Remaining fission gas and fission product inventories measured by gamma spectrometry on the sample.

5. Key dates

MERARG is currently operational for LWR fuels. After a complementary safety assessment of the facility to take into account the SFR fuels conditions, it could be used for SFR fuels tests.

6. Location

CEA, Cadarache centre, France.

CEA-VERDON hot cell facility

1. Project title

New hot cell facilities devoted to the study of irradiated fuel behaviour and fission product releases under simulated accident conditions.

2. Objective

To address the major safety issue relative to the fission product (FP) release and transport during a hypothetical severe accident occurring in a nuclear power plant (NPP).

3. Relevance to deploying SFR prototype and potential benefits

VERDON circuits exhibit two complementary configurations: one devoted to FP releases from the fuel, and the other to study their transport in the primary system of a NPP. In the future, the VERDON facility could also be used to qualify fuels for GEN-IV reactors.

4. Infrastructure needs

The VERDON laboratory is mainly composed of two hot cells and one glove box located within the STAR facility at the CEA Cadarache Centre. The first hot cell is dedicated to sample preparation and storage, as well as all pre- and post-test examinations.

The second hot cell is specifically devoted to the VERDON experiments and contains two complementary experimental circuits to study a) the FP release tests using an aerosol filter, and b) the FP transport using four sequential thermal gradient tubes.

Facility main features

- Sequence carried out on a pellet or a short pellet stack.
- Temperature time history up to fuel melting.
- Possible test atmospheres: He, H_2, H_2O, mixed H_2/H_2O, air, O_2/H_2.
- On-line measurement of fission gas and fission product release (sample, May-pack, filter, gas trap).

5. Key dates

VERDON will be operated first on PWR fuels in 2011-2013. Available after refurbishment for potential other fuels not before 2014 (to be discussed) MOX, carbide or metal fuels for sodium fast reactors (SFR).

6. Location

CEA, Cadarache, France

VERDON: overview of the cell equipment and focus on the furnace

Facilities at the *Institut de radioprotection et de sûreté nucléaire (IRSN)*, France

The CABRI experimental reactor facility for SFR R&D studies relative to fuel safety and to severe accidents issues

The CABRI experimental reactor (operated by CEA at Cadarache Centre) is dedicated to fuel safety experiments designed by IRSN within the framework of its R&D programmes.

1. Main characteristics

The CABRI facility has been intensively used in support to SFR R&D studies on fuel accident behaviour during the period 1973-2001.

The CABRI facility allows in-pile simulation of, for instance, incidental and accidental SFR transients (ULOF, slow and fast UTOPs). Its main characteristics are the following:

- core power: 20 MW (transient) with cosine axial power profile;
- core fissile height of 80 cm, close to SFR reactor core height: this allows the direct use of SFR irradiated pins without any refabrication (different from PWR rods testing);
- test section located in the core center: single pin or multi-pin (3) testing device;
- power transients starting from reactor nominal conditions (400-600 w/cm) and generated by He3 valves depressurisation (15 b) which allows versatile transient power shape (various power injection kinetics);
- energy deposition inside the fuel pin evaluated within ±5%, as a result of the thermal balance (steady-state runs);
- thermal-hydraulics conditions (in sodium loop conditions): Tinlet~400°C, Toutlet = 600°C max., Na velocity: 4-5 m/s;
- possible simulation of loss of flow followed by a power transient triggered at a given time.

2. Measurement devices

Inside the test section	Inside the reactor facility
• Fluid temperature at different axial levels. • Inlet and outlet flow rates, channel voiding. • Channel pressures below and above test pin. • Acoustic devices for pin failure detection. • Fuel and clad elongation (transient).	• Hodoscope measurement (neutron detector focused on test pin) allowing kinetics and fuel motion quantification (molten fuel in the channel) and precise determination of final state of the pin degradation. • Pre and post-test examinations (NDE): X-ray radiography, tomography, gamma spectrometry (destructive examinations to be performed in hot cells).

3. Relevance for SFR fuel safety and severe accidents issues

For all types of anticipated fuel types and burn-up levels:

- irradiated fuel behaviour under slow power transient conditions: margin to fuel melting, margin to deterministic pin failure;
- irradiated fuel behaviour and associated phenomena under severe accidents such as unprotected loss of flow or local blockage, fast or slow power transients.

4. Infrastructure needs, availability

The facility has been renewed for an additional operating period of 40 years. A pressurised water loop has been implemented and the CIP programme under OECD auspices is scheduled from 2011 to 2014. Prospective studies are being launched at IRSN in view of:

- improvement of the driver core performance for wide range of transients simulation (i.e., increased energy injection capability);
- implementation of an integrated sodium loop devoted to SFR tests while keeping the capability of performance of pressurised water loop tests.

Depending on the feasibility and work performance, the facility may be available for testing from 2020.

5. Location

CEA, Cadarache centre, France.

Facilities in Germany, various institutions

Experimental facilities for sodium fast reactor (SFR)

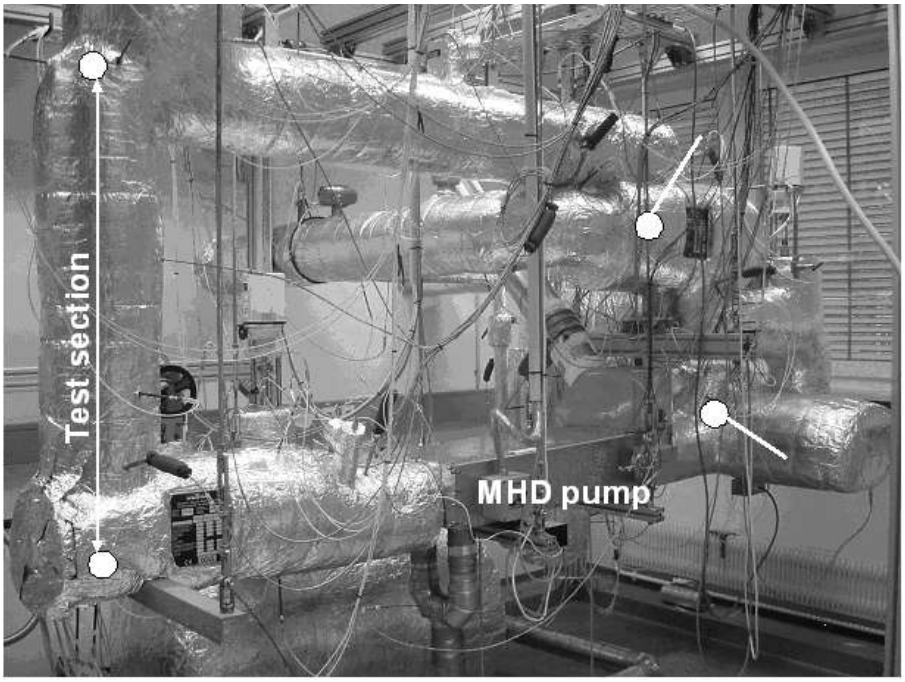
ALINA – Karlsruhe Institute of Technology (KIT)

The ALINA (***a**nlage für **li**thium und **na**trium*) facility is part of the **ka**rlsruhe **l**ead **la**boratory (KALLA). The oil-cooled ALINA loop operates with sodium and is related to windowless targets with free surfaces as well as to small-scale generic heat transfer experiments. It also allows to investigate corrosion in flowing sodium.

Typical characteristics are a temperature range from 150 to 400°C, max. flow rate of app 21 m^3/h, heating power up to 120 kW, and a system pressure of up to 3.5 bar. ALINA has one test port with a height of 1.5 m. The sodium inventory amounts to 150 litres. Since commissioning in 2007, it has operated for approximately 3 000 h.

Contact: stieglitz@iket.fzk.de

Corrosion test loop ALINA



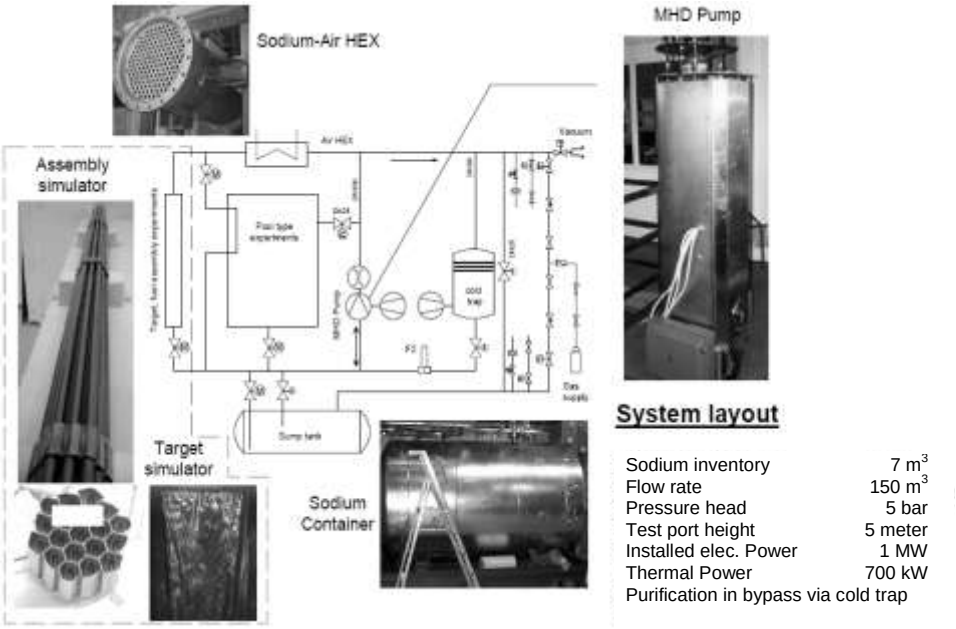
KASOLA – Karlsruhe Institute of Technology (KIT)

The **KA**rlsruhe **SO**dium **LA**boratory (to be erected in 2010/11) is to investigate thermal hydraulics of fuel assembly simulators at prototypical height and power level heat transfer studies Nu = f (P/D, spacer, design, flow induced vibrations, flow rate) as well as evaluation of flow pattern in pool configurations of reactors and operational characteristics (first stage: slab configuration). The simulation will include loss of heat sink scenarios as well as loss of flow scenarios.

- Improvement of adequate thermal-hydraulic scale coupling methods in reactor design codes for fast systems.
- Validation and evaluation of uncertainty threshold of system codes for fast reactor applications in nominal and abnormal operation scenarios.
- Validation and modification of turbulent liquid metal heat transfer models in CFD tools on limited geometric scale.
- Development of models to describe free surface liquid metal flow.
- Development of free surface liquid metal targets for nuclear applications.
- Investigation and development of in-service inspection and repair monitors for Na-systems.

Contact: stieglitz@kit.edu; hering@kit.edu

Schematics of the KASOLA facility



Liquid sodium facility NATAN – Forschungszentrum Dresden/Rossendorf (FZD)

A liquid sodium facility working with about 100 litres is readily available at the Institute of Safety Research at *Forschungszentrum Dresden-Rossendorf*. Working temperatures are up to 300°C. An electromagnetic pump circulates the melt with velocities of up to 2 m/s in pipes of typically 50 mm in diameter. Various test sections exist. In order to study gas bubble two-phase flows, argon can be injected into the flowing sodium.

The facility is equipped with several measuring techniques, allowing in particular the measurement of the Na velocity. Integral flow rate measurements are done with a new type of electromagnetic, fully contactless flow rate sensor. The resolution of local velocity profiles over the cross-section of a pipe is possible by means of ultrasonic Doppler velocimetry. This technique even works through the wall of the pipe.

A new type of heat exchanger has been installed in an available lead loop, but could also be installed in the sodium facility. It makes use of the liquid metal, GaInSn, as an intermediate medium between the hot melt and the cooling water. In this way, the possibility could be significantly reduced for a direct contact between the hot sodium and the cooling water in case of some leakage.

Contact: g.gerbeth@fzd.de

Facilities in India, Indira Ghandi Centre of Atomic Research (IGCAR)

Fast breeder test reactor

1. Description

The fast breeder test reactor (FBTR) has been in operation since 1985. It was built with French collaboration, and the design closely follows that of Rapsodie-Fortissimo, with the major difference that its tertiary loop is a steam-water system with four steam generator modules, as against the sodium-air heat exchanger in Rapsodie. FBTR uses a unique, high Pu mono-carbide fuel as the driver fuel.

The reactor went critical with a small core of 23 fuel sub-assemblies rated for 10.6 MWt. The excellent performance of the carbide fuel resulted in the extension of the burn-up limit of this fuel to 155 GWd/t from an initial conservative design limit of 25 GWd/t. This extension was possible through PIE of the fuel at 25, 50 and 100 GWd/t and modelling by the fuel designers. The current limit of 155 GWd/t is based on the residual ductility of the wrapper. The extension of the burn-up limit resulted in progressive expansion of the core to compensate for the reactivity loss due to burn-up. The current core has 50 fuel sub-assemblies of different compositions-viz. 70%PuC+30%UC, 55%PuC+45%UC and MOX with 44% PuO_2. The current core has a power rating of 18.6 MWt.

The current major mission of FBTR is to test irradiate MOX fuel of power reactor composition (29%PuO_2) to a target burn-up of 100 GWd/t at a LHR of 450 W/cm. Operating experience with the sodium systems has been excellent, and the four sodium pumps have cumulatively logged 600 000 h of trouble-free operation. The steam generators have operated without any leaks.

In addition to routine measurements of control rod worths and reactivity coefficients of inlet temperature and power after every core configuration change, physics experiments carried out are: reactor kinetics experiments, void coefficient measurements, response of delayed neutron detection system to detect clad failure, and flux mapping in sodium above core. A series of safety related engineering tests was conducted in 1994-95, basically to validate the codes used in incident analysis. Normal plant incidents like off-site power failure and tripping of one pump in the primary, secondary or tertiary loops were studied and the sequence of events confirmed to be as per safety logic, and the temperature transients in components were recorded. Primary pump coast down characteristics, take over by the batteries and low speed running of the pumps were studied. As a precursor to the station black out test, natural convection tests in the secondary and primary loops were separately carried out. In order to validate the performance of the delayed neutron detection (DND) system and verify the capability to identify any failed fuel, a series of experiments was conducted with a special sub-assembly with 19 perforated pins of natural uranium. The results of the experiments were satisfactory, and confirmed that the failed fuel could be easily detected by the DND system and the failed fuel location could be easily inferred from the contrast ratio of the counts from both the loops.

2. Inside reactor/outside reactor

This is the reactor facility.

3. Instrumentation

Peak flux: $2.65*10^{15}$ n/cm^2/s for a fresh core. Depending upon the core burn-up, it may go to $3.15*10^{15}$ n/cm^2/s.

Irradiation in capsules within steel reflector and Nickel reflector sub-assemblies (capsules of size 19 mm to 25 mm) or special fuel sub-assemblies with seven pins removed.

Capsule size: 10 mm ID, 550 mm long.

Sample temperature: 380ºC to 490ºC.

4. Current status and availability

Presently the reactor is operated in campaign mode. Depending upon the irradiation needs and programme, operation is planned. Reactor is currently available.

5. Uniqueness

This facility has employed a unique plutonium rich carbide fuel which has reached a burn-up of 165 GWd/t without any fuel pin failure so far. It has played a key role in the finalisation of India's 500 MWe PFBR design.

500 kW sodium loop

The 500 kW sodium loop was designed and constructed mainly to test the performance of the indigenously-developed components such as the sodium-to-sodium heat exchanger, sodium to air heat exchanger, sodium immersion heaters, etc., and their associated instrumentation. This loop was commissioned in 1976. This facility is located in Engineering Hall-I.

1. Details of the facility

Sodium inventory of this facility is 3.5 m^3. A heater vessel (HV) with immersion heaters of 500 kW capacity is the heat source and a sodium-to-air heat exchanger (NaX) of 500 kW capacity is the heat sink. A figure of eight configuration with HV on hot leg and NaX on cold leg, 100 m^3/h capacity centrifugal pump for sodium circulation, on-line purification system and associated pipe lines are the main components of this facility.

2. Operating range

1. Maximum operating temperature: 823 K
2. Maximum flow: 100 m^3/h at 0.4 MPa
3. Material of construction: AISI 304
4. Purity of sodium: reactor grade

3. Type of testing

Dynamic testing in sodium and can be used for experiments requiring high flow conditions.

4. Instrumentation

- *Temperature measurement*: K-type thermocouples wired to control panels, monitored and controlled through data acquisition system and PID controllers.
- *Leak detection system*: wire type, spark plug type conductivity based detectors wired to electronics of leak detection system in instrument panel.

- *Sodium level detection system*: sodium level monitored through continuous mutual inductance type level sensors and discontinuous resistance type level sensors wired to control panel through electronics of level detection system to control instrument panel.
- *Sodium flow measurement*: permanent magnet type flowmeters at required locations wired to flow indicators in instrument panel.
- *Control and safety logics*: hard wired relay based electronic components from individual sensors to the control panel with wall mimic.

5. Capability of the facility

This facility was constructed mainly to test the performance of the indigenously developed components like sodium to-sodium heat exchanger (IHX), sodium to air heat exchanger (NaX), EM pumps, sodium immersion heaters, etc. The other experiments conducted were testing of eddy current flow meter (ECFM), centrifugal pump, calibration of eddy current flow meter to be used in PFBR primary pump, PM flow meters, etc.

6. Current status

In operation.

7. Availability and further programmes

Testing and development of instruments for void detection related to gas entrainment in primary circuit and calibration of eddy current flowmeters to be used in PFBR are planned in this facility.

8. Uniqueness

Oldest test facility with a centrifugal pump utilised for sodium circulation. Apart from FBTR this is the only test facility with a centrifugal pump in operation. This loop can be utilised for experiments with high sodium flow requirements.

Large component test rig (LCTR)

The large component test rig (LCTR) was constructed in Engineering Hall-III of IGCAR to carry out full scale testing of few critical components of the prototype fast breeder reactor (PFBR) in sodium under simulated reactor operating conditions.

1. Details of the facility

Sodium inventory of the facility is 90 m^3. There are three test vessels in which independent test conditions can be maintained. Two storage tanks, online purification system with air cooled cold trap, EM pump (FLIP) for sodium circulation, heater vessel with immersion heaters are available in this test facility.

2. Operating range

1.	Maximum operating temperature:	823 K
2.	Maximum flow:	20 m^3/h at 0.56 MPa
3.	Material of construction:	AISI 316
4.	Purity of sodium:	reactor grade

3. Type of testing

This loop can be utilised for dynamic and static testing in sodium.

4. Instrumentation

Same as the 500 kW loop.

5. Capability of the facility

Three large test vessels form part of this test facility. One test vessel is of 7 m^3 sodium hold up, the second test vessel is of 47 m^3 sodium hold up and third test vessel is of 17 m^3 sodium hold up.

Full scale testing of prototype components of PFBR is being conducted in this test facility. Some of the important tests done in this test facility are:

- performance testing of sodium to air heat exchanger;
- heat transfer and temperature distribution studies of roof slab, rotatable plug and control plug model of PFBR;
- performance testing and qualification of shut down mechanisms such as control safety rod drive mechanism (CSRDM) which was qualified for 18 years of reactor operations and diverse safety rod drive mechanism (DSRDM) which was qualified for 10 years of reactor operations;
- performance testing and qualification of a sodium vapour condenser to be used in the cover gas purification circuit which will be utilised in case of failure of fuel pins;
- testing of ultrasonic under sodium scanner as part of developments related to in-service inspection of PFBR.

Sodium testing of 20 kW capacity immersion U heaters, testing of annular linear induction pump, calibration of PM flowmeters/mutual induction type continuous level probes, testing of DC conduction pump were taken up as part of development activities.

6. Current status

In operation.

7. Availability and further programmes

- Qualification of CSRDM & DSRDM for 40 years.
- Testing and qualification of fuel transfer mechanisms for PFBR.
- Testing and qualification of failed fuel location module for PFBR.

8. Uniqueness

Large sodium test facility and full scale testing of prototype components of PFBR have been conducted and can be utilised for similar activities. Since there are three test vessel simultaneous testing schedules are possible.

Sodium water reaction test facility (SOWART)

A study on self-wastage and impingement wastage is necessary for the safe operation of the steam generator. In the sodium water reaction test facility (SOWART), a steam leak is simulated and the sodium water reaction is being carried out. This facility was located in Engineering Hall-III.

1. Details of the facility

Sodium inventory of this facility is 10 tons. Facility consists of an on-line purification system with an air cooled cold trap, EM pump (flat linear induction pump) for sodium circulation, two heater vessels with immersion heaters, steam injection system, hydrogen sensors, micro leak test section (MLTS) and impingement wastage test section (IWTS).

2. Operating range

1. Maximum operating temperature: 803 K
2. Maximum flow: 20 m^3/h at 0.37 MPa
3. Material of construction: AISI 316
4. Purity of sodium: reactor grade

3. Type of testing

This loop can be utilised for dynamic testing of components in sodium apart from the sodium water reaction studies.

4. Instrumentation

Same instrumentation along with the hydrogen measurement indicated below.

5. Hydrogen meters

- Measurement of dissolved hydrogen through in sodium sputter ion pump based hydrogen meter and electro-chemical type hydrogen meters.

- Measurement of hydrogen gas using sputter ion pump based hydrogen meter and thermal conductivity detector.

6. Capability of the facility

This facility was constructed to study the behaviour of self wastage and leak enlargement during sodium-water reaction, to study impingement wastage and to develop leak detection methods. Self wastage studies were carried out at different steam leak rate in the range of 10-50 mg/s. Model cold trap testing was completed in this test facility. Different hydrogen meters were calibrated by injecting known quantity of hydrogen.

7. Current status

In operation.

8. Availability and further programmes

Studies on a large number of SG tube material specimens for adjacent tube wastage studies, self wastage studies and development, and testing of in-sodium and cover gas hydrogen meters are planned for this facility.

9. Uniqueness

Only test facility in IGCAR where sodium water reaction experiments are conducted.

LEENA facility

It is required to assess the performance of the wire-type leak detectors and confirm that they are meeting the requirements. A test facility by the name of LEENA was constructed to test the wire-type leak detector lay out by simulating actual sodium leaks of different rates. This facility is located in Engineering Hall-III.

1. Details of the facility

The sodium inventory of this facility is 300 litres. The LEENA facility consists of a sodium storage tank, a test vessel and test sections. There are three different test sections simulating different horizontal and vertical sodium pipelines of PFBR secondary circuit. Separate leak simulators are provided on each test sections for creating sodium leaks.

2. Operating range

1. Maximum operating temperature: 873 K
2. Maximum flow: Static test facility
3. Material of construction: AISI 316

3. Type of testing

Qualification of sodium leak detector layout for different pipe sizes.

4. Instrumentation

Similar to other sodium loops.

5. Capability of the facility

This facility was constructed for the performance evaluation of the wire-type leak detector layout of the PFBR secondary circuit by creating sodium leaks. Different experiments were conducted in different test sections at 350 and 550°C. The lowest leak rate that can be simulated is 214 g/h.

6. Current status

In operation.

7. Further programmes

Lower leak rate experiments and testing of sandwich leak detectors are planned for this facility.

8. Uniqueness

This test facility was exclusively set up to study and qualify the leak detector layout of PFBR. Developmental activities on different types of leak detection mechanisms are possible in this loop.

SADHANA facility

A 355 kW capacity sodium test facility named SADHANA (SAfety Grade Decay Heat removAl loop in Natrium) was constructed at IGCAR to study the thermal hydraulics behaviour of safety grade

decay heat removal of PFBR. This facility is located in Engineering Hall-III. In SADHANA the sodium in Test Vessel 4 (TV 4) which simulates the hot pool of the PFBR is heated by immersion electrical heaters. This heat is transferred to the secondary sodium through DHX. The secondary sodium is circulated in the secondary loop by the buoyancy head developed in the loop due to the temperature difference in hot and cold legs of the loop. The heat from the secondary sodium circuit is rejected to the atmosphere through the AHX. A 20 m high chimney develops the air flow required to transfer the heat from the secondary sodium to the atmosphere through the AHX.

1. Details of the facility

The capacity of SADHANA loop is 355 kW and the height difference between the thermal centers of DHX and AHX is 19.5 m. This 1:22 scaled model loop is designed on Richardson number similitude. The sodium-hold up in this facility is 3 m^3. The experimental facility contains a sodium-to-sodium decay heat exchanger, sodium-to-air heat exchanger, a test vessel containing sodium pool, chimney, and associated piping. An annular linear induction pump (ALIP) is used for sodium circulation in the primary side. Immersion heaters are used for heating the sodium pool in test vessel.

2. Operating range

1. Maximum operating temperature: 873 K
2. Maximum flow: 5 m^3/h at 0.5 MPa
3. Material of construction: AISI 316 L
4. Purity of sodium: reactor grade

3. Type of testing

Safety grade decay heat removal experiments with different types of heat exchangers.

4. Instrumentation

Similar to other sodium loops.

5. Capability of the facility

This facility was constructed to study the thermal hydraulic behaviour of SGDHR system. The facility was commissioned and has completed around 500 hours of high-temperature operation. At 550°C, the sodium pool temperature in the test vessel and the secondary sodium loop generated a sodium flow of 6.7 m^3/h and approximately 420 kW of heat power.

6. Current status

In operation.

7. Availability and further programmes

Heat removal capacity of DHX and AHX at different sodium temperatures and sodium levels, measurement of air flow distribution in AHX shell side, behaviour of SGDHR during SCRAM and behaviour of SGDHR during station blackout are planned in this facility.

8. Uniqueness

Constructed to understand and prove the decay heat removal capability of SGDHR system exclusively to study the natural circulation behaviour in the secondary loop. It is the first of its kind in India.

Other installations

SOFI facility

The SOFI facility was built for the experimental simulation of molten fuel coolant interactions after a core disruptive accident which involves melting of the grid plate by the molten fuel, heat transfer between dislocated molten fuel with surrounding sodium, solidification of molten fuel followed by fragmentation, settlement behaviour of core debris on the core catcher and heat transfer from the debris surface to the sodium pool. These simulations will be carried out in various phases. In the first phase, simulants like woods metal will be used in a water system. In the second phase, molten 304 L and sodium are used, and in the third phase molten UO_2 and sodium will be used. For simulating decay heat, heating coils will be provided on the surface of the core catcher. Imaging of the core debris dispersion pattern on the core catcher for sodium experiments is being developed. In the test during phases 2 and 3, core debris movement will be visualised.

Sodium fire facilities (MINA and SFTF)

Basically, we have two facilities for sodium fire studies. The first facility, called MINA, is for simulating sodium fire (pool as well as spray fire), involving a few grams of sodium (spray) and a few tens of kilograms of sodium (pool) to understand the basic science of a sodium fire. The facility consists of a rectangular steel chamber (6m x 6m x 5m) which has integrated sodium loops for injecting the required amount of sodium. It is provided with state-of-art instrumentation for measuring the relevant parameters. The second facility, the sodium fire test facility (SFTF), is for performing the sodium fire studies on a larger scale involving quantities up to 500 kg (mixed spray and pool fire). This facility will be used for qualifying sodium leak collection trays, sodium fire fighting system, sodium concrete interaction and sodium aerosol behaviour in a larger volume.

Facilities in Italy

LIFUS5 (ENEA)

The facility, located at ENEA-Brasimone research centre, is designed to study the interaction at high temperatures and in a wide range of conditions between heavy liquid metal (LBE, lead) and steam/water, which may occur in a fast breeder reactor intermediate system in the event of a steam generator tube rupture. In case of facility refurbishment, it is also possible to investigate the interaction of heavy liquid metal with sodium following a heat exchanger tube rupture event.

1. Facility description

With reference to the LIFUS5 synoptics (http://web.brasimone.enea.it), the facility mainly consists of:

- a reaction vessel S1 (volume = 0.1 m^3; design temperature = 500°C; design pressure = 20 MPa), where the interaction takes place. S1 contains U-shaped cooling tubes mock-up in one of the four sectors in which the vessel is divided by two plates. A quantity of water ranging between 1.0-6 Kg, which can be injected in 5-8 s at a pressure of 15.5 MPa and a temperature ranging between 265 and 325°C (LIFUS5 can operate also at different operating conditions);
- a pressurised water vessel S2 (volume = 0.015 m^3; design temperature = 350°C; design pressure = 20 MPa), containing water to be injected in S1 at a fixed pressure;

- an expansion vessel S5 (volume = 10.1 l; design temperature = 500°C; design pressure = 20 MPa), connected to the reaction vessel through four expansion tubes, one per sector;
- a dump tank S3 (volume = 2 m^3; design temperature = 400°C; design pressure = 1 MPa) to collect the gaseous and aerosol interaction products from S1 at the end of the test or, if the system pressurisation exceeds 190 bar, the steam discharged from S1-S5 by means of the rupture disk D1 opening;
- a liquid metal storage tank S4, where the metal is molten for filling the reaction vessel S1 and the expansion vessel S5 by means of an electromechanical pump.

The test parameters (pressure, temperature, etc.) are acquired by a fast data acquisition system with a dedicated software in LABVIEW environment. The test instrumentation consists of:

- 8 water-cooled high precision piezometric fast pressure transducers;
- 18 K-type quick response thermocouples;
- a differential pressure transducer for measuring the S2 water level; and
- 2 level meters to measure the liquid metal level in S1 and S5. Moreover accelerometers may be placed on the mock-up tubes in order to study the propagation and the effects of the pressure waves and, in particular, to verify the integrity of the tubes placed just close to the rupture position.

2. Facility operation

All operations of facility control and supervision are carried out from the control room by means of a PLC system integrated with a PC that allows the synoptic display of the facility as well as to set in each component the control and alarm parameters (temperature, pressure, liquid metal and water level).

3. Availability

Experimental campaigns in support to the GEN-IV sodium-cooled reactors are possible during the periods in which the facility is not used for fusion and other testing activities.

4. References

http://web.brasimone.enea.it/ then select “experimental activities” ⇒ main facilities ⇒ liquid metal loops ⇒ LIFUS5
A. Ciampichetti *et al.* (2008), “LBE-water Interaction in Subcritical Reactors: First Experimental and Modelling Results”, *J. of Nuclear Materials 376,* pp. 418-423.

Facilities at the Japan Atomic Energy Agency (JAEA), Japan

Experimental facilities for sodium fast reactor (SFR)

Joyo – Japan Atomic Energy Agency (JAEA)

Joyo is an experimental fast reactor, which attained initial criticality in 1977 as the first LMFR in Japan. From 1983 to 2000, Joyo operated with the MK-II core as an irradiation test bed to develop the fuels and materials of fast reactors. The reactor was upgraded to the MK-III core in 2003, increasing reactor power and hence the irradiation performance.

In addition to the un-instrumented fuel irradiation sub-assemblies, Joyo has various irradiation rigs with temperature control, flow controlled irradiation, in-core low temperature test loop and ex-vessel irradiation rig with load control system. Especially, the material testing rig with temperature control (MARICO) can control the irradiation temperature within a precision of 4°C during the irradiation period.

Joyo has been stopped since June 2007 due to the mechanical trouble in handling the test rigs, which damaged the upper internal structure. Investigations on the range of damage and the cause of the trouble and the repairing plan are underway.

Typical characteristics of Joyo are the inlet/outlet temperature of 350/500°C, total thermal power of 140 MW, core height/diameter of about 0.5/0.8 m and maximum neutron total/fast (E>0.1 MeV) flux of $5.7*10^{15}/4.0*10^{15}$ n/cm^2/s.

Contact: tobita.yoshiharu@jaea.go.jp
www.jaea.go.jp/04/o-arai/joyo/indexes.htm
www.iaea.org/inisnkm/nkm/aws/fnss/fulltext/te_1405_2.pdf

Sodium safety facilities – Japan Atomic Energy Agency (JAEA)

The **TRUST-2** facility simulates the high-temperature rupture of steam generator pipe in prototypical condition. The test section consists of test pipes which contains high pressure/temperature water at controlled flow rate, which is heated electrically from outside. The capacities of water heater are the volume of 75 itre, maximum pressure/temperature of 21 MPa and 450°C. The maximum heating power is 500 kW. Recent activities were made to measure the transient heat flux in the water side under rapid heating condition and to obtain mechanical properties of the steam generator pipe for the JSFR plant.

SWAT-1R aims at investigating the water-sodium interaction in the pipe rupture accident in steam generator. The characteristics of the reaction tank are the volume of 630 litres, maximum pressure of 1.96 MPa and maximum temperature of 580°C. The same water heater used with the TRUST-2 facility is used for SWAT-1R. The temperature distribution of the sodium-water reaction jet, heat transfer coefficient between the jet and pipe structure and the final amount of wastage were measured mainly by thermocouple signals. The future test programme will concentrate on the self-wastage behaviour of double-tubed steam generator pipe in JSFR.

SWAT-3R is a large-scale test facility that aims to investigate the propagation of a steam generator tube rupture in prototypical condition. The amount of sodium in the sodium loop is 15 ton, its rated temperature is 555°C, the maximum pressure is 1.96 MPa and the maximum flow rate is 2.3 m^3/min. The volume of the reaction tank is 10m^3. The SWAT-3R facility has two water heaters with their volume of 3.1 and 4.8 m^3, the maximum pressure of 24 MPa and maximum temperature of 425°C. The measurement is made mainly by thermocouple signals and system pressure in the reaction tank. Preparation for the future test is under way to investigate the basic experimental knowledge on the behaviour of reacting jet at medium leak rate range.

SAPFIRE is a test platform to investigate the various phenomena in a sodium leak accident such as spray, columnar and pool-type fire, sodium-concrete interaction, sodium-structure interaction with chemical reaction and aerosol behaviour. The volume of the test section is about 100 m^3, allowable pressure is 0.2 MPa and the maximum ventilation rate is 70 Nm^3/min. The total amount of sodium in the test loop is 10 tons.

MELT – Japan Atomic Energy Agency (JAEA)

The MELT facility was built to perform out-of-pile experiments relating to the molten material behaviour in-core disruptive accident of a fast reactor. The main part of this facility is an induction-heating crucible capable of generating a melt of alumina, steel and tin, with the highest temperature of 2 300°C and maximum amount of 20 litres. The power of induction heating is 300 kW. This facility had been utilised in the investigation of fuel-coolant interaction (FCI) in water system and the structure erosion behaviour by melts jet. A small sodium loop to perform FCI experiments in sodium system is also available. X-ray video and high-speed video are available for the visualisation of multiphase transient phenomena in the test section.

PLANDTL – Japan Atomic Energy Agency (JAEA)

1. Objective

The PLANDTL is a plant dynamics testing loop for sodium. It was mainly developed to perform thermal hydraulic experiments to develop the components in fast breeder reactors. The main targets of the investigation are:

- thermal transient behaviour in decay heat removal after scram;
- heat removal characteristics in natural convection; and
- thermal striping behaviour.

2. Main characteristics

The PLANDTL facility consists of primary and secondary sodium loops coupled thermally at the IHX module. The characteristics of primary/secondary loops are: maximum sodium flow rate of 1 200/600 litre/min., design temperature of 650°C and total amount of sodium is around 10 tons. The primary loop is connected to a test section, which simulates the thermal-hydraulics in reactor vessel. This reactor vessel section has 6 fuel sub-assemblies with 37 fuel pins. Each fuel pin is heated electrically with chopped cosine power distribution along 1 m. The pin diameter is 8.3 mm or 20.8 mm. The peak heat flux is 2 MW/m^2 and allowable temperature is 950°C. The sodium flow rate to each sub-assemblies and also inter-wrapper gap can be controlled independently. The outlets of sub-assemblies are connected to upper sodium plenum. The heat balance and facility size are shown in the table below. The heat generated in the test section is transferred to the secondary loop and cooled by an air cooler. The upper plenum and the upper part of IHX have heat exchangers to simulate the decay heat removal system. Measurements are made by thermocouples located to more than 860 points in the test loop and electromagnetic flow meter.

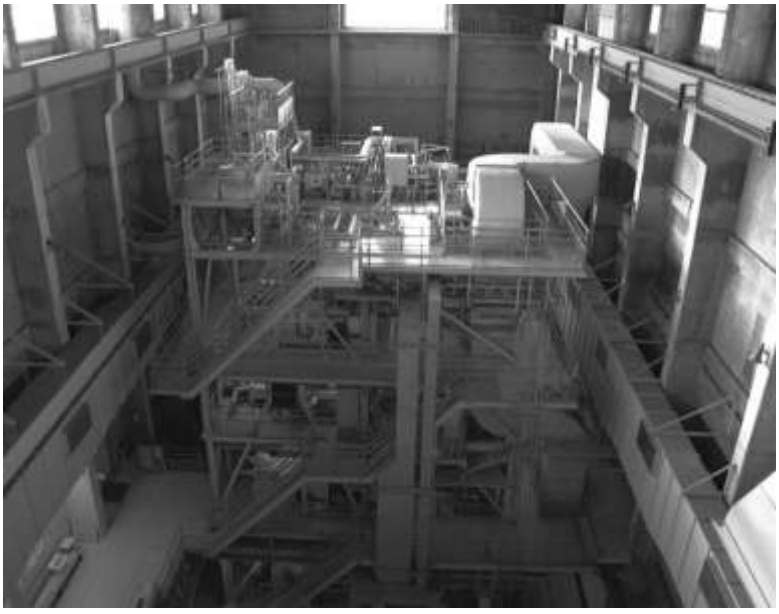
3. Current status

The current activity of PLANDTL is concentrated on the thermal-hydraulics of DHX and PRACS in JSFR.

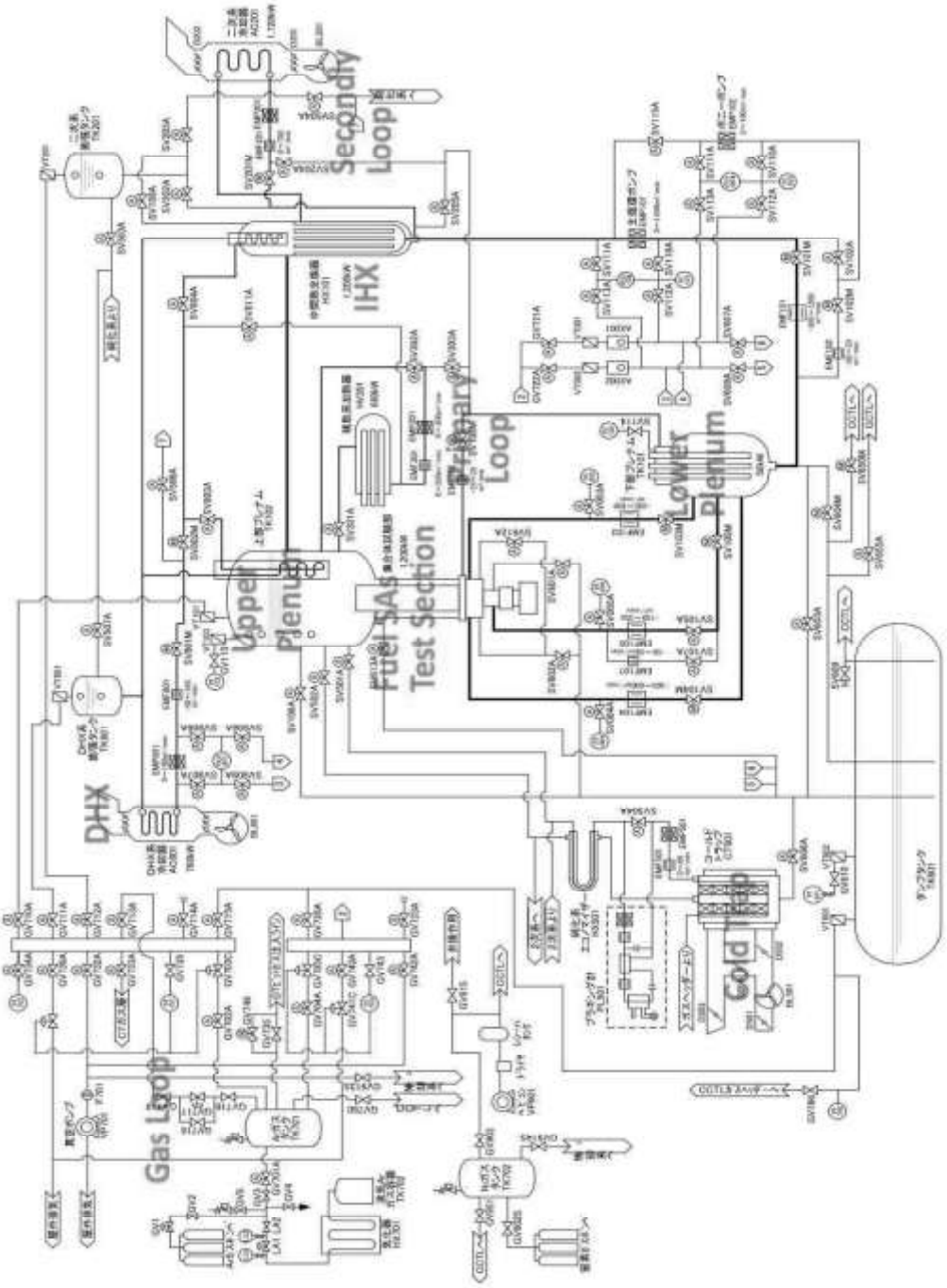
Main characteristics of PLANDTL

Heat balance	
Pin bundle	1 200 kW
Auxiliary heater	680 kW
Total	1 720 kW
IHX	1 720 kW heat removal
Air cooler	1 720 kW heat removal
DRACS	150 kW heat removal
Facility size	
Main pipe diameter	4 inch
Core length	2.31 m (inlet: 0.92 m, active heater: 1 m, exit 0.39 m)
Upper plenum	2 m ID, 2.6 m sodium level
Sodium inventory	9 ton
Maximum pressure	0.8 MPa
Maximum temperature	625°C
Flow rate	1.2 m^3/min

Bird view of PLANDTL loop



Structure of loop and components of PLANDTL



TTS – Japan Atomic Energy Agency (JAEA)

1. Objective

Investigation of thermal fatigue behaviour of the piping and components in SFR plants is the main purpose of TTS. TTS realises the transient temperature fluctuation by mixing low and high temperature sodium flow and observes the formation and growth of cracks.

2. *Main characteristics*

The PLANDTL consists of the high temperature loop, the low temperature loop and the mixing test section. The mixing of high and low temperature sodium is controlled to realise a wide variation in the temperature fluctuation pattern. The sodium out flow from the test section is lead to the mixing vessel to accommodate the temperature fluctuation and returned to the high and low temperature loop.

The structure around the test section is also exposed to temperature fluctuation, and therefore special attention is paid to sodium leakage from the test loop by providing a leakage tray, drain pipe, sodium leakage detector and video cameras.

3. *Current status*

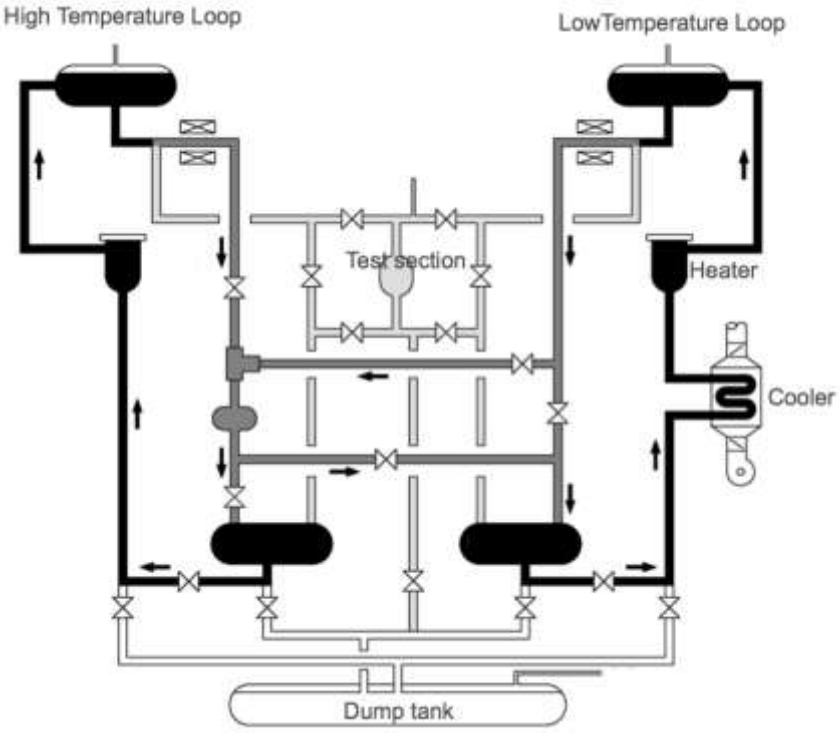
The current activity of TTS is concentrated on the thermal fatigue phenomena in JSFR.

Main characteristics of TTS

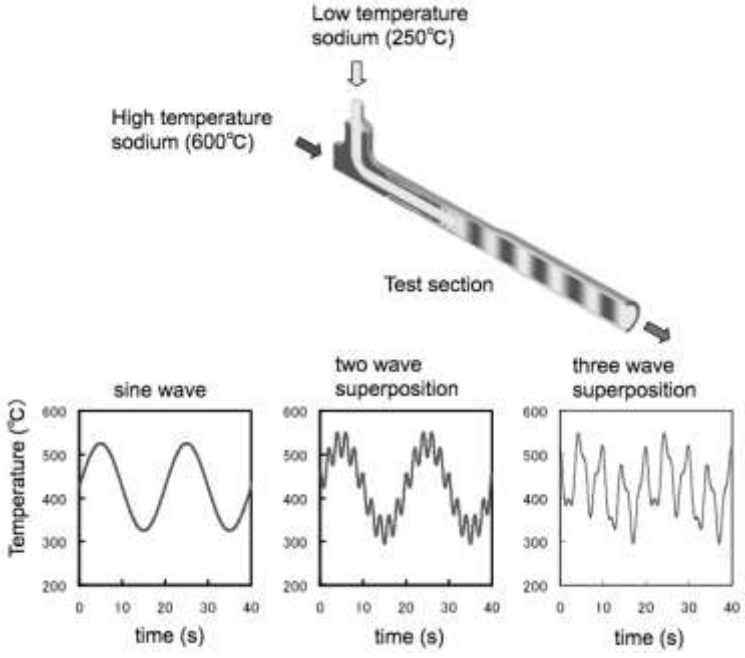
Heat balance	
Main heater	1 180 kW
Auxiliary heater	680 kW
Total	1 720 kW
Facility size	
Main pipe diameter	3 inch
Core length	2.31 m (inlet: 0.92 m, active heater: 1 m, exit 0.39 m)
Sodium inventory	76.8 ton
Maximum pressure	0.8 MPa
Maximum temperature	650°C
Flow rate	0.4 m^3/min

View of TTS loop

Structure of loop and components of TTS



Conceptual figure of temperature fluctuation in TTS



AtheNa – Japan Atomic Energy Agency (JAEA)

The large-scale sodium testing loop complex (AtheNa) is under planning development. The construction of this facility will begin soon and the first operation is scheduled to be in 2012. This facility consists of several sodium test loops. The mother loop supplies sodium at 250°C to each testing loop. The sodium pot facility has three sodium pots, varying their diameter from about 1.5 to 5 m and height from about 5 to 15 m depending on the objectives of tests. The amount of sodium in these pots ranges from about 20 to 100 m^3 and the temperature range covers from 250 to 550°C.

The complex has a sodium heater by LPG of 60 MW, which is connected to SG testing loop and system testing loop. SG testing loop has a SG with straight double-walled tube with a capacity of about 8 MW. The flow rate of primary sodium loop is around 2.4 m^3/min. The system testing loop simulates the overall cooling system in JSFR consisting of IHX, SG and water system. The heat capacity of this loop is about 55 MW.

The test loop for the development of pump-integrated IHX supports a sodium flow rate of about 32 m^3/min and the pipe diameter is 16 B. A full-scale IHX will be installed in this loop.

Facilities at Korea Institute of Atomic Energy (KAERI), Korea

For the verification of the full passive decay heat removal system (PDRC), a large-scale sodium thermal-hydraulic test is required to demonstrate the cooling capability of the PDRC during the long- and short-term periods after reactor shutdown. The preliminary design of the test facility has been made recently using the KALIMER-600 as a reference reactor, and its basic design will resume in 2011 based on the demonstration reactor which is intended to be constructed by 2028 according to the Korean long-term SFR development plan. The installation of the facility is scheduled to be completed by the end of 2013. The main experiments will commence from 2015 after the start-up test in 2014.

The major design characteristics of the preliminarily designed facility are as follows.

Reference reactor	KALIMER-600
Height and length scale	1/5
Volume scale	1/125
Aspect ratio scale	1/1
Pressure and temperature	1/1 (prototypic condition)
Working fluid	Sodium (17 ton)
Simulated core (electrical heater) power	7% of the scaled nominal power (1.9 MW)
Design code for major component	ASME Sec. VIII Division 2
Main test section	PHTS and PDRC with same configuration as KALIMER-600

Currently, the scoping analyses to assess the appropriateness of the design methodology are on-going using the system code and the CFD method. Also, the separate experimental works to verify some of the scaling principles adopted in the facility design are being performed. For the visual investigation of multi-dimensional flow and temperature distributions inside reactor vessel, 1/10 scale model for the reactor vessel of KALIMER-600 is being constructed using plexiglass with the plan of starting the test soon. The test loop for assessing the performances of IHX, AHX, DHX and primary pump is being made to demonstrate the design characteristics and verify the component design codes.

Facilities in the United States

Annular core research reactor (ACRR)

1. Objective

The annular core research reactor (ACRR – water pool type) is used for radiation effects studies and reactor safety experiments. Experimental capabilities include fuel transient testing under abnormal and/or accident conditions for SFRs, LWRs, and GCRs. The flexible power shaping allows for fast power transients from zero initial power, slower transients from an arbitrary power level, and steady-state conditions such as those needed for post-accident heat removal studies.

2. Relevance to deploying SFR prototype and potential benefits

The ACRR has been used extensively to perform nuclear heating tests on SFR, LWR, and GCR fuels (both intact and degraded core geometries). Typical experiments conducted in ACRR have included partial length fuel assembly transient testing, post-accident heat removal from debris beds, fuel failure phenomenological tests, margin-to-failure tests, fuel/clad interactions, and equation-of-state tests. The ACRR can perform programmed transient or steady-state (up to 4 MW) heating to simulate specific reactor conditions of interest. Sodium cooled experiments have been and can be conducted in the ACRR. Experiments in the ACRR were fundamental for developing the equation of state for SFR fuel as well as establishing the database for critical heat flux in debris beds.

3. Infrastructure needs

The ACRR is a water-moderated, natural convection cooled, pool-type reactor fuelled with UO_2-BeO ceramic fuel pellets in stainless steel cladding. The fuel elements are arranged in a tight-pitched hexagonal lattice core around a dry central irradiation cavity. The unique 9-inch diameter dry central cavity allows for installation of experimental packages containing coolant loops, nuclear material, and advanced diagnostics. Here, experiments are driven by the epithermal neutron spectrum (the average neutron energy is 0.66 MeV) of the core over its approximately 20-inch active fuel height. Test assemblies can include stagnant capsules, a single fuel rod in flowing coolant, or multiple rods in flowing coolant. Within the open central cavity, instrumentation and control cabling can remain connected to packages during irradiation operations, allowing for real-time monitoring or control of items being irradiated. The reactor sits near the bottom of a 8.6-m-deep tank with a diameter of 3.1 m. The ACRR has a diameter of about 1 m, and it is offset to one side in the tank. Thus, a large region exists next to the reactor where larger experiments can be placed. The facility is highly flexible and suited for numerous uses.

4. Key dates

Available and fully operational. Major upgrades completed in the last decade.

5. Location

SNL Albuquerque, NM, United States.

Core alloy flow and erosion (CAFE)

1. Objective

The CAFE facility is a hands-on laboratory facility for generating kilogram-quantity melts of low-radiological-hazard metallic nuclear fuel materials (e.g., un-irradiated uranium-iron alloy) and discharging them into various receiver geometries such as, tubes and troughs.

2. Relevance to deploying SFR prototype and potential benefits

Chemical interactions between metallic fuels and cladding, and also between metallic fuels and structural materials, have long been an important part of the assessment of cladding failure and post-failure motions, and relocations of fuel-bearing melts during postulated SFR severe accidents. In systems in which the fuel and cladding have high uranium and iron contents, respectively, margins to cladding failure are strongly dependent upon the generation and interaction between low-melting-point uranium-iron compositions and the formation of U-Fe compounds. Other fuel and cladding constituent elements also play a role in that interaction. Upon cladding breach, molten fuel moving within the cladding and exiting the cladding breach is thus typically in the form of a non-homogeneous, possibly two-phase, mixture of fuel and cladding alloys. After leaving the cladding, that mixture may chemically interact with, ablate, and absorb material from the structure along which it flows, resulting in changes in its physical, chemical, and thermal properties. Understanding and modelling such behaviour is a key aspect of ensuring that SFR reactors will provide safe, in-vessel retention of molten core material during severe accidents.

3. Infrastructure needs

The CAFE apparatus provides a vacuum, induction heating, and melting of materials contained within suitable crucibles. After the melt has reached the desired temperature, a plug in the bottom of the crucible is removed to allow the melt to drain from the crucible, penetrate a thin diaphragm that is part of the vacuum boundary, and enter the user-designed structure that will receive the melt. The 1.4 m x 1.2 m x 0.6 m inerted containment below the induction heating chamber includes mounts for three video cameras which have been used to monitor melt flow down open, inclined troughs.

4. Key dates

The facility is currently operational.

5. Location

Argonne National Laboratory, Illinois, United States.

CYBL

1. Objective

Large-scale, natural convection testing.

2. Relevance to deploying SFR prototype and potential benefits

The cylindrical boiling facility (CYBL) is in Technical Area III at the Sandia National Laboratories (SNL) Thermal Test Complex (TTC) and simulates a full-scale nuclear reactor vessel for studying advanced safety concepts; especially natural circulation. It has the capability to provide an environment and instrumentation for natural convection testing and would be a very suitable facility for construction of a sodium test. The facility is currently being used to experimentally simulate BWR

and PWR spent fuel natural convection cooling, heating, fuel ballooning, and ignition. The cavity is an ideal containment for sodium loops due to the stainless steel and has 5MW of power for electrical heating.

3. Infrastructure needs

The CYBL facility was originally built for studying phenomena for boiling process from a downward-facing torispherical surface. The test apparatus was originally a tank-within-a-tank design. The replaceable inner test vessel, simulating the reactor vessel, is 316 L stainless steel, 3.7 m in diameter and 6.8 m high. The outer tank simulating the reactor cavity is also made of 316 L and is 5.1 m inner diameter and 8.4 m high. It has been reconfigured to use the outer tank as a dry containment for experiments involving full scale PWR and BWR fuel assemblies. The CYBL test apparatus is installed over an observation pit. There are 51 viewing windows, ranging in size from 0.3 m to 0.6 m in diameter, on the side and bottom of the outer vessel for observing the boiling process. Arrays of thermocouples are used to measure the fuel assembly surface temperatures and a Class 3/4 laser system monitors the convective thermal-hydraulic characteristics during the test. All instrumentation is complemented with a fully qualified data acquisition system.

4. Key dates

Currently conducting fuel meltdown experimental studies for LWRs.

5. Location

SNL Albuquerque, NM, United States.

Melt coolability and concrete interaction (MCCI)

1. Objective

The primary focus of the MCCI test facility for the last ten years has been the conduct of large scale integral- and separate-effect reactor material tests that address ex-vessel severe accident core melt stabilisation issues for both current as well as advanced (Gen 3+) light water reactor (LWR) plant designs.

2. Relevance to deploying SFR prototype and potential benefits

The MCCI test facility is currently being utilised for the conduct of large scale ex-vessel severe accident experiments that involve a high degree of hazards including the use of radioactive materials, steam explosion and hydrogen production issues. Thus, the cell provides an excellent working environment for the conduct of experiments with significant hazards which might be relevant to SFR such as sodium-concrete, sodium-water, or fuel-coolant interaction tests.

3. Infrastructure needs

The MCCI facility consists of a large (1 000 m^3) reactor test cell that was designed for experiments involving high hazards (i.e., the cell has an explosion rating of 45 kg TNT equivalent, and a static pressure rating of 4 bar). In the current tests, large core melt masses (up to 2 metric tons) are produced at an initial temperature of ~ 2 100ºC using exothermic chemical mixtures over a timescale of ~ 30 seconds. Once produced, the melts are resistance heated (up to 500 kW) to simulate decay heat. The melts are generated within test sections that mock up generic features of current as well as advanced plant designs. Typically, the melt begins erosion into a highly instrumented concrete crucible; the interaction is then flooded from above to determine the ability of water to cool and stabilise the interaction. In other tests focused on advanced plant concepts, the effectiveness of cooling

and stabilising the melt from below with water is investigated. Steam explosion and hydrogen production are the primary safety issue associated with these types of tests, and the above specifications (i.e., explosion and overpressure ratings) define the overall safety envelope within which the tests are conducted.

4. Key dates

Operational and available, with a test scheduled in 2010.

5. Location

ANL Argonne, IL, United States.

NSTF

1. Objective

The NSTF is an existing large-scale natural convection heat transfer test facility that is currently being refurbished as part of the NGNP programme to provide code validation and overall performance data for the passive reactor cavity cooling system (RCCS) designs that have been proposed for both the pebble bed and prismatic gas reactor concepts.

2. Relevance to deploying SFR prototype and potential benefits

The NSTF was originally developed to provide code validation and overall performance data for the radiant vessel auxiliary cooling system (RVACS) of the General Electric PRISM SFR design that was developed under DOE sponsorship as part of the ALMR programme. The facility was also used to provide performance data for the reactor auxiliary cooling system (RACS) for the Rockwell SAFR SFR design. As noted above, the facility is currently being refurbished as part of the NGNP programme to provide test data for the evaluation of various RCCS designs that are proposed for both prismatic and pebble bed gas reactor concepts. Aside from heat flux and temperature distributions on the exterior of the reactor, the heat transfer performance of the RCCS/RVACS is essentially decoupled from details of the reactor design (e.g., sodium vs. gas-cooled). Thus, the experiment capabilities of the NSTF are deemed to be highly relevant to the SFR, particularly for small fast reactor designs in which exterior cooling of the reactor is relied upon as the primary method for achieving shutdown heat removal.

3. Infrastructure needs

The NSTF is a large-scale (26 m tall), highly instrumented test facility for providing natural convection heat transfer data for RCCS/RVACS design concepts. The working area that mocks up the region around the reactor vessel is geometrically flexible so that different RCCS cavity designs can be accommodated; i.e., spacing between the heated and unheated cavity walls can be varied from 30 to 150 cm; the cavity width is 132 cm. The overall height of the heated zone is 6.7 m. The heater assemblies can provide fluxes up to 23.7 kW/m^2; operating temperatures can range up to 677 ºC. The facility can be run in either temperature- or heat-flux control modes along the 6.7 m heated length, with the capability to provide axial power shaping. Designs are currently being developed for placing generic mockups of air-cooled RCCS components within the working area of the test facility for evaluation testing.

4. Key dates

Under refurbishment; facility will be operational mid-2011 assuming funding is allocated.

5. Location

ANL Argonne, IL, United States.

Sodium technology and advanced component test facility

1. Objective

To provide a flexible test bed and required infrastructure necessary for testing and evaluating advanced SFR components and technologies, including sodium compatibility of advanced materials.

2. Relevance to deploying SFR prototype and potential benefits

A number of advanced components, sodium technology advancements, and advanced materials have been identified or proposed for improving SFR efficiency and reliability, thus reducing SFR costs, since serious sodium technology development was abandoned some decades ago. The subject facility has the flexibility to test these advanced components, sodium technology advancements, and advanced materials at the 3-5 cubic meter test apparatus scale. Presently, the facility houses a loop for characterising sodium plugging phenomena that are of concern in advanced printed circuit heat exchangers (PCHE) currently being investigated as the IHX for supercritical CO_2 Brayton cycle BOPs. The combined PCHE SCO_2 BOP would be more compact, of higher efficiency, and would reduce the hazards of primary Na/secondary fluid interactions. A second loop is comprised of a specimen exposure vessel, an electromagnetic pump, two electromagnetic flow meters, an economiser, a heat exchanger and a cold trap. The objective of this loop is to provide an initial assessment of sodium corrosion compatibility of advanced materials being considered for sodium reactor applications. In addition, results from this loop will also provide valuable insight into licensing and regulation issues associated with materials compatibility (such as coolant impurity limits) that were raised during evaluation of the Clinch River breeder reactor and initial reviews of the PRISM design. A new containment structure is under construction for accommodating additional component testing of advanced sodium pumps, the Na/CO_2 interaction, freezing/thawing of Na within components (especially the PCHE), and several advanced instrumentation candidates.

3. Infrastructure needs

The facility is well supported by its own infrastructure, in addition to two on-site alkali metal scrubbing facilities for component cleaning. One of the scrubber booths can handle radioactive waste.

4. Key dates

Work is active and ongoing.

5. Location

ANL, Chicago, IL, United States.

SURTSEY

1. Objective

The SURTSEY facility has been used for LWR severe accident studies, including FCI and DCH. SURTSEY can be used for direct atmospheric heating and large tests involving molten materials and their interactions. SURTSEY is currently in use to generate data for sodium fire phenomena model development.

2. Relevance to deploying SFR prototype and potential benefits

Within the past year, several molten sodium experiments including pool fires and spray fires have been successfully performed in both open (outdoor) configurations and within the SURTSEY pressure vessel. Available facilities include a pressurised sodium melt generator, a range of instrumentation including high-speed photo-metrics, and a complete data acquisition system. Complementary advanced fire modeling development activities are also ongoing.

These same facilities could easily be adapted to investigate sodium-concrete, sodium-water, and sodium-coolant interactions. These types of experiments would only require relatively minor modifications to the existing test facilities and would benefit from the existing capability to melt and deliver sodium in either a pour or spray form.

3. Infrastructure needs

The SURTSEY facility consists of a pressure vessel oriented vertically with the lower head flange approximately two meters above the ground. Instrumentation includes various pressure and temperature measurements to track the dispersal of debris. Additionally, the SURTSEY facility allows for debris and aerosol material to be sampled and recovered. Current sodium fire testing is using state-of-the-art diagnostics to characterise particle size distributions, temperatures, and heat fluxes.

4. Key dates

The SURTSEY facility is operational and flexible. Any required modifications to adapt to specific test requirements could be implemented in less than one year depending on the nature and scale of testing desired.

5. Location

SNL Albuquerque, NM, United States.

Transient reactor test facility (TREAT, graphite reactor)

1. Objective

An air cooled irradiation test facility designed to evaluate reactor fuels and structural materials under conditions simulating various modes in nuclear reactors.

2. Relevance to deploying SFR prototype and potential benefits

Possible slow power transients under special conditions designed to test the behaviour of various fuels and structural materials under accidental transient conditions of SFRs (ULOF, TOP) starting from zero power. TREAT has been used to test irradiated prototypic length, oxide and metal fuel under transient conditions (e.g., fuel from FFTF, EBR II and PFR), and it could also be used to test minor actinide fuel elements under similar transient conditions. This reactor played a very important role in examining transient fuel pin behaviour. It helped to identify negative feedback associated with transient behaviour. The axial locations of fuel failure in both metal and oxide fuel leading to dispersal and shutdown were also seen in TREAT. Phenomena such as fuel freezing, Na fuel interactions, eutectic formation were also discovered in TREAT.

3. Infrastructure needs

The TREAT facility is an air-cooled, thermal, heterogeneous test reactor with a 1.22 meter core height. The design allows experiments to replace one or more of the 10.2 x 10.2 cm fuel elements in

various parts of the core (usually the centre). The usual test assemblies consist either of stagnant capsules, the single pin in flowing sodium test loop, or the multiple pin (up to seven) in flowing sodium Mark III loop. The TREAT core acts as a driver for the experiment. Very severe accidents, including those with fuel failure and consequent fuel coolant interaction, can be simulated. TREAT can be used to expose fuel to extreme conditions so that the failure limits may be investigated and safety factors established. In addition, the fuel damage limits determined in TREAT can be used to assess other reactor designs for safety during upset conditions. TREAT is equipped with a fast neutron hodoscope capable of determining fuel motion in an experiment under severe accident conditions. Hence, not only is the failure limit determined, but the failure mode and degree of failure can be observed.

4. Key dates

Shut down; decision for restart expected in 2010.

5. Location

INL Idaho, United States.

Under sodium viewing (USV) test facility

1. Objective

This bench-scale liquid sodium test facility is designed for development and testing of ultrasonic under-sodium viewing systems that are needed for in-service inspection and repair of a sodium-cooled fast reactor (SFR).

2. Relevance to deploying SFR prototype and potential benefits

The USV test facility provides the typical SFR environment for testing and evaluating high-temperature ultrasonic transducer and ultrasonic imaging system. Thus, the facility provides an integrated approach to developing and demonstrating key transducer technologies needed for in-service inspection, supporting maintenance activities, and on-line monitoring as applied to SFR technology.

3. Infrastructure needs

The USV facility consists of mainly two pressure vessels: one, called the test tank, is used for under-sodium viewing tests and the other is a sodium dump tank. Both tanks are connected to a vacuum pump line and an argon gas supply line. The two tanks are connected through a ½" stainless steel tube with Swagelok VCR face seal fitting. The tube facilitates sodium transfer between the two tanks. Both tanks are placed inside drip pans. The test tank was built from an 8.0" schedule 40 stainless steel 304 (SS 304) pipe to which two 8.0" SS 304 schedule 40 caps were welded, one on both ends. The length of the tank is 20". Each access port on top of the vessel has an opening of 3.07", to which a 5.0" pipe is welded on, and a 3" slip-on flange class 300 is welded onto the pipe. The dump tank was constructed in the same way as the test tank. But, the total length of the dump tank is 16". A maximum of 2.8 gallons of sodium can be placed in the dump tank. Both tanks were built and tested according to ASME Code Section VIII. Sodium level, temperature, and pressure sensors are provided to each tank for process control. The two ports on top of the test tank are designated for target sample and test transducer. The target sample port allows for the insertion of the target assembly that consists of a mounting rod and a sample holder to which the target sample is attached. The mounting rod will provide the Z-axis adjustment while the sample holder will position the target directly underneath the test transducer. A graphite washer is used to seal the cover flange of the sample port. To facilitate the

scanning capability, the transducer assembly that goes into the transducer port is firmly attached to a scanning stage that is sandwiched in between two graphite washers. The translation stage provides movements in X- and Y-directions. The image scan and processing are controlled by LabView and MatLab.

4. Key dates

Available.

5. Location

ANL, Lemont, IL, United States.

Appendix 2

TERMS OF REFERENCE (T.O.R.) OF THE TAREF TASK

T.O.R. for the CSNI Task on Advanced Reactor Experimental Facilities (TAREF)

REVISION of 4 November 2008

Title	**Task on Advanced Reactor Experimental Facilities (TAREF)** **Infrastructure for safety research (focus on gas reactors and sodium fast reactors)**
Objective	The objectives of this activity are as follows: o Provide an overview of identified facilities (existing or planned in TAREF member countries) suitable for performing safety research investigations relevant to gas cooled reactors (GCR) and sodium fast reactors (SFR) o Review the phenomenon identification and ranking tables (PIRT) that have already been carried out for gas reactors in order to identify important safety issues o Identify safety issues relevant to sodium fast reactor (recognising different designs) o Propose a strategy for an efficient utilisation of facility and resources for meeting short- and long-term safety requirements unique to GCR and SFR
Scope	The tasks to be performed are as follows: o Compile a questionnaire regarding: a) facilities that are suitable for experimental studies on gas cooled reactor safety; b) facilities that are suitable for experimental studies on sodium-cooled reactor safety o Examine the PIRT that are available for gas reactor systems[3] and, based on the questionnaire responses and on the PIRT outcome, evaluate the options that can be recommended to CSNI for facility utilisation in safety domains through international undertakings. This is envisaged to require one or two meetings of the sub-task dedicated to gas reactors o Identify the issues that are relevant to sodium reactor safety. This is expected to require three meetings of the sub-task dedicated to sodium reactors o Based on the questionnaire responses and on the safety issue identification performed for sodium reactors, evaluate the options that can be recommended to CSNI for facility utilisation in safety domains through international undertakings. This is envisaged to require one additional meeting of the sub-task dedicated to sodium reactors.

3. The USNRC has recently conducted a PIRT on gas reactor systems, and will make it available to the Task Group.

Safety significance	Advanced reactors incorporate design features, materials and safety provisions that are likely to require exploratory experiments, verifications and confirmatory tests. For this, an adequate facility and expertise infrastructure will be needed, in support of safety evaluation. A strategy for an optimal development and utilisation of such infrastructure is key for producing the necessary data in a timely and efficient manner, as required for safety assessments.
Expected outputs	Two reports (to CSNI and CNRA), one for gas reactors and one for sodium reactors, identifying: a) facilities relevant to safety research on identified safety issues; b) recommendations on strategy for facility and expertise utilisation, e.g., on facilities and programmes needed at international level in support of safety assessments.
Safety issues and topics covered	o New concepts of operation o New risk perspective and safety requirements o Fuel and fuel cycle safety o New materials and fabrication technologies o Experimental facility loss (or need, in this case) o Transparent technical basis for safety assessment
Schedule/milestones	o Develop the questionnaires and receive responses in the first half of 2008 o On gas reactor facilities, produce an outline of the report content by end of 2008 and the final report by June 2009 o For sodium reactor facilities, identify relevant safety issues and produce the final report by June 2010
Lead organisation	o USNRC for gas reactor systems o IRSN for sodium reactor systems
Participant organisations	Countries having advanced reactor programmes or large facilities suitable for gas reactor and sodium reactor studies, or countries that can contribute to the discussions on advanced reactor safety assessments are expected to participate.
Resources	o Approximately 1.5-2 man-years for the part related to gas reactors o Approximately 3-3.5 man-years for the part related to sodium reactors
Interaction with others	o WGAMA on fluid dynamics and accident issues o WGFS on fuel issues o WGIAGE on structural materials issues
Approval by CSNI	Approved at the CSNI meeting in December 2007

Appendix 3

SUMMARY OF THE TAREF MEETINGS ON SFRS

First meeting of the TAREF Task Group

OECD Headquarters, Paris, 3-5 November 2008

Summary (SFRs)

1. The terms of reference of the TAREF group were revised. Changes did not alter the substance of the task, except for what concerns the definition of the issues for sodium reactors, which would be based on discussions within the group rather than on a PIRT. The modified text was circulated at the meeting and is in the CD-Rom than was distributed at the meeting. The NEA Secretariat will circulate the modified text to the CSNI PRG and the Bureau.

2. The reports (one for GCR and one for SFR) will be organised in four chapters as follows:

   EXECUTIVE SUMMARY

   1. INTRODUCTION

   2. OVERVIEW OF GCR DESIGNS (or SFR DESIGNS)

   3. TECHNICAL ISSUES AND ASSOCIATED FACILITIES

      - Technical areas
      - Issues pertaining each technical area
      - Facilities vs. issues

   4. CONCLUSIONS AND RECOMMENDATIONS

      a. Graphite and ceramics
      b. TRISO fuel

3. For SFR, the following technical areas were provisionally identified:

   a. Thermal-hydraulics
   b. Fuel safety
   c. Reactor physics
   d. Severe accidents
   e. Integrity of structures
   f. Others

4. Regarding fuel safety, the USNRC will clarify what is meant with SFR fuel handling as a safety concern (as indicated in the DOE questionnaire answer).

5. In preparation of the next meeting, France (Ms. Papin and Mr. Renault) and Japan (Mr. Tobita) will provide an initial list of issues + description associated to the above provisional technical areas, by end of January 2009. This will constitute the basis for the discussion that is to take place at the next meeting to finalise the technical areas and related issues.

6. For each of the GCR and SFR facilities proposed, members should provide a ½ page text (no figures) providing the information that the proposer believes is most appropriate (in- or out-of-reactor, operating range description of the test section, type of testing, instrumentation, current status and availability, uniqueness, links to appropriate web site). Members should provide this information by end of January 2009 (independent from point 7). In doing this, contributors should help addressing reasonable size facilities, considering grouping for clusters of small facilities.

7. The next TAREF meeting is scheduled for 2-4 March 2009 in Paris, starting at noon of the first day and finishing at noon of the last day. A small pre-meeting might be considered for the morning of the 2nd March. The main focus of the meeting would be:

    a. For SFR, in-depth discussion on the Issues/Description (see Point 10) and description of facilities proposed (Point 11).

    b. For GCR, review of the material assembled for Chapters 1, 2 and 3 (Points 2 to 7), review the description of facilities (Point 11) and discussion on conclusions and recommendations.

Considering the schedule for reporting to CSNI, a small follow-on meeting might be considered for the GCR, possibly in conjunction with the CSNI-PRG meeting (28-29 April 2009).

Agenda of the first meeting

General

1. Opening
2. Election of Task Chairpersons
3. Adoption of the agenda
4. Discussion and approval of Terms of Reference
5. Overall scope and intended schedule
6. Group output, content of final report
7. Working plans for group meetings to complete work scope – GCRs and SFRs

Gas reactors

8. Overview of safety research in TAREF countries
9. Overview of PIRT
10. Overview of questionnaire
11. Facilities for resolution of safety issues (current and planned)
12. Approach to set priority
13. Working plan and preparations for next meeting

Sodium reactors

14. Overview of safety research in TAREF countries
15. Approach to definition of safety issues

- PIRT approach
- SFEAR approach

16. Overview of questionnaire
17. Facilities for resolution of safety issues
    - Current
    - Planned
18. Working plan and preparations for next meeting
19. Closure, date and place of next meeting

Second meeting of the TAREF Task Group

OECD Headquarters, Paris, 2-4 March 2009

Summary (SFR part)

1. All action items from the previous meeting have been satisfactorily addressed. In particular, members have been very active in providing the requested material. Some of the information received needed some adjustment/completion, which to an appreciable extent was done during the meeting. It was agreed that the available information was adequate for the purpose of completing the task on GCR and for preparing the SFR discussion.
2. As agreed at the previous meeting, the intended schedule is to complete the GCR report in time for the CSNI June 2009 meeting. The SFR report should be finalised one year later, i.e., in June 2010. The main purpose of the meeting was to develop near, medium- and long-term recommendations for CSNI on GCR, and to discuss and bring forward the set of safety issues related to SFR.

SFR task

3. It was confirmed that the SFR report will have the same structure of the GCR report, as well as the same approach, i.e., develop technical areas, safety issues in each area and aggregate facilities to each safety issue.
4. *Chapter 1, Introduction:* It was agreed that Chapter 1 would be basically the same as for the GCR.
5. *Chapter 2, Outline of SFR systems:* It was agreed that France (Mr. Chalaye, CEA) and Japan (Mr. Tobita, JAEA) will contribute with a description of the two respective systems of interest, i.e.,
    a. The pool type reactor (CEA) and loop type reactor (JAEA).
    b. CEA (Mr. Chalaye) will also include a very concise description of more innovative SFR concept (or concepts) if appropriate, as well as a description of the fuel types. The US is to help informing on past experience with metallic fuel.
    c. The effort should be proportioned to the expected size, which totals ~10 pages for Chapter 2. To be sent to members before end of August 2009.
6. *Chapter 3, Technical areas, issues, facilities:* The focus of the meeting was the discussion of the safety issue, which was centred around the list of issues and description that had been

communicated by IRSN before the meeting. The discussion pointed out some aspects that need to be taken into account in the follow-on of the SFR task. In particular:

a. The IRSN list and description is a good start. Where feasible, the safety issues should be better identified, possibly splitting some of the issues in more specific issues.
b. In some cases, a description of existing knowledge based on past experience is given, in other cases not. It is suggested to have the existing knowledge mentioned for all issues, as this can help evidencing what is not available and needs assessment.
c. The attempt made at the meeting to rank the issues in terms of H, M, L scores for safety relevance and state of knowledge, revealed that the score may be very dependent on SFR reactor concepts, which may be to a great extent not yet defined. Some thinking on best way forward is needed.
d. The SFR Chair will in preparation of the next meeting address the above considerations and prepare a revised list of issues and description, including if appropriate a first indication of priority (H, M, L for importance and knowledge). To be sent to members by end of July 2009.

7. The NEA Secretariat will send out a request for facility descriptions that might be missing for the appendix. Members will also be requested to provide a table of facility vs. issues (using the current list of issues). This will be done in March 2009.

8. The next meeting should be a two-day meeting and is to be held at the NEA headquarters between the end of September-beginning of October 2009 period. The NEA is to propose a date based on conference room availability.
(Additional note: booking has been made for a conference room at the NEA headquarters, NOT where we were for the previous meeting, for the week 28 September-2 October. There is no availability on the following week).

Agenda of the second meeting

General

1. Opening
2. Adoption of the agenda
3. Review of summary and actions from the previous meeting
4. Expected outcome of the meeting

Gas reactors

5. Recall of final report structure and Task member contributions
6. Status of Chapter 1: Introduction
7. Status of Chapter 2: Overview of GCR designs
8. Status of Chapter 3: Technical issues and associated facilities
   - Descriptions of the technical areas
   - Descriptions of the issues pertaining to each technical area

- Facilities vs. issues

9. Priority setting
10. Next steps, tasks and schedule
11. Members' ideas/suggestions for international undertakings

Sodium reactors

12. Recall of (provisional) technical areas: are changes needed?
13. In-depth discussion of technical issues in each area (SFEAR approach)
14. Way to finalise technical issues, tasks
15. Discussion on facilities (*based on partners information)*, relation to issues
16. Priority setting
17. Structure of final report, schedule
18. Tasks allocation to produce the final report
19. Next steps, tasks and schedule
20. Members' ideas/ suggestions for international undertakings
21. Working plan and preparation of the next meeting
22. Closure, date and place of the next meeting

Third meeting of the TAREF Task Group

NEA Headquarters, Issy-les-Moulineaux, 29 September–1 October 2009

Summary

The third TAREF meeting was entirely dedicated to SFR topics as the GCR report is being issued.

Its main objectives were to exchange and to discuss information in order to finalise the PIRT-like approach related to sodium fast reactor (SFR) safety issues, to rank the relevant experimental facilities and to prepare the final report. In particular, the tasks were:

- to review and discuss the Chapter 1 (Introduction) and Chapter 2 (Outline of SFR systems);
- to review and discuss the safety issues in each technical area in terms of description and ranking (importance level and knowledge level);
- to review and discuss the tables of facilities versus safety issues, based on information provided by partners; and
- to establish a table of facility ranking and priority setting (see Appendix 3).

On the basis of the documents prepared and sent prior to the meeting, of the presentations (JOYO, JHR, IGR capabilities) and from the discussion, the TAREF Task Group members:

1) agreed on the proposed table describing each safety issue and its ranking (importance level and knowledge level);
2) fulfilled a table including, for each issue, experimental facilities with their ranking score taking into account their relevance to the issue, their uniqueness, the readiness of the experimental team and their cost; and
3) proposed to distinguish among the experimental facilities those available within 3 years, those available between 4 and 8 years and those which will be available in the longer term (more than 8 years).

This approach firstly underlined that no experimental reactor is available in the short term for addressing irradiated fuel accidental behaviour for SFR (see Appendix 3).

In order to extend the basis of experimental facilities addressing a maximum number of safety issues, the TAREF members also agreed to add a chapter dedicated to relevant Russian experimental facilities provided that appropriate information will be made available.

Furthermore, the participants could agree about the next steps towards the completion of TAREF mandate, including the next TAREF meeting which will be held in Paris, on 2-3 February 2010 and the deadline for the draft of the final report by 15 March 2010, for submission to the PRG.

Summary of actions

In the following sections, the actions resulting from the discussions will be summarised:

1. Chapter 1 (Introduction) was reviewed and revised, in particular with suppression of "and licensing purposes" at the end of the first paragraph in Section 1.2. The German representatives requested to mention in the minutes their wish to keep the initial wording although they accept the majority decision. The NEA Secretariat will circulate the modified text to the TAREF members by end of October 2009.
2. Chapter 2 (Outline of SFR systems) is missing information on metallic fuel. The USNRC will provide such information by end of October 2009.
3. Chapter 2 was reviewed in detail. Several modifications were proposed; in particular, it was recommended to avoid advertising one design to the detriment of others. The NEA Secretariat will revise Chapter 2 by incorporating the comments and information on metallic fuel and will submit it to Messrs. Chalaye (CEA) and Tobita (JAEA) who should finalise and circulate Chapter 2 to TAREF members by end of November 2009.
4. Chapter 3 dedicated to the technical areas, safety issues and experimental facilities will include an introduction of approximately one page for each of the technical areas and a set of safety issues for each technical area. These issues correspond to those agreed by TAREF members to be of high importance/low to medium knowledge. Each issue should be accompanied with a short description explaining the phenomena involved and the safety implication. An indication can be given as to whether there is adequate ongoing work covering the issue in question. The Chair (IRSN) agreed to draft a proposal to be circulated to the TAREF members by end of November 2009.
5. The chapter related to the Russian experimental facilities will be centralised by the NEA Secretariat. TAREF members are encouraged to contribute by sending available information to the NEA Secretariat. Depending on the availability of the information, decision on keeping this chapter in the report will be taken early January 2010.

6. For each of the SFR facilities proposed, members should provide a half of a page of text (no figures) giving the information that the proposer believes is most appropriate (in- or out-of-reactor, operating range description of the test section, type of testing, instrumentation, current status and availability, uniqueness, links to appropriate web site). Members should provide this information by mid-November 2009. In doing this, contributors should help addressing reasonable size facilities.
7. Information related to IGR and to TTS should be added and information on some CEA facilities should be complemented, all of them to be provided in the same way by mid-November 2009.
8. Daniel Magallon will circulate information on FRAG facility by end of October 2009 (already sent to the Chair).
9. Decision was taken to exclude MONJU from the list of potential facilities addressing SFR safety issues.
10. For the safety issue E5 (sodium leak evolution and detection), Wolfgang Hering proposed to use experience feedback to define an experiment. He will check if his institute (INR) can propose an experiment to address this issue and will inform the TAREF members by end of October 2009.
11. The Chair and the Secretary will prepare by 15 December 2009 a proposal on highly ranked facilities for the short, medium and long terms, including the rationale of selection.
12. The 4th TAREF meeting is scheduled for 2-3 February 2010 in Paris, starting at 2 p.m. on the first day.

Agenda of the third meeting

Day 1 ***(Tuesday, 29 September 2009)***

14:00	Opening and welcome	Chair
14:15	Approval of the agenda	Secretariat
14:30	Approval of the summary record of the 2nd TAREF meeting – Review of actions from the previous meeting	Chair + Secretariat
15:00	Expected outcome of the meeting	Chair
15:15	Recall of approach, structure of final report, deadline	Chair
15:30	*Coffee break*	All
15:45	Review of Chapter 2 (outline of SFR systems)	All
17:00	Chapter 3: Recall of technical areas and in-depth discussion of technical issues in each technical area (SFEAR approach): issue description, ranking.	All
17:50	Summary of the first day	Chair
18:00	End of day 1	Chair

Day 2 ***(Wednesday, 30 September 2009)***

09:00	Chapter 3: Recall of technical areas and in-depth discussion of technical issues in each technical area (SFEAR approach): issue description, ranking (Cont.)	All
10:30	*Coffee break*	All
11:00	Chapter 3: Recall of technical areas and in-depth discussion of technical issues in each technical area (SFEAR approach): issue description, ranking (Cont.)	All
12:30 13:30	*Lunch break*	All
13:30	Chapter 3: Discussion on facilities (based on partners information) in relation to the technical issues • *Review of tables "Facilities vs issues" – Discussion* • *Tables for ranking* • *Priority setting*	All
17:20	Summary of the second day	Chair
17:30	End of day 2	Chair

Day 3 ***(Thursday, 1 October 2009)***

09:00	Review of Chapter 1	All
10:00	Tasks allocation to produce the final report	All
10:30	*Coffee break*	All
10:45	Next steps: tasks and schedule	All
11:15	Members ideas/suggestions for international undertakings	All
11:35	Working plan and preparation of the next meeting	All
11:50	Date and place of the next meeting	All
12:00	Closure of the meeting	Chair

Fourth meeting of the TAREF Task Group

NEA Headquarters, Issy-Les-Moulineaux, 2-3 February 2010

Summary

The main objectives of the fourth TAREF meeting were to exchange and to discuss information in order to finalise the PIRT-like approach related to sodium fast reactor (SFR) safety issues, to rank the relevant experimental facilities and to prepare the final report. In particular, the tasks were:

- to review and endorse the revised Chapter 1 (Introduction) and Chapter 2 (Outline of SFR systems);
- to review and discuss Chapter 3 (technical areas, safety issues and facilities);
- to draw and to discuss preliminary conclusions and recommendations.

On the basis of the documents prepared and distributed prior to or during the meeting, and from the discussion, the TAREF Task Group members:

1) endorsed Chapter 1 and agreed to endorse Chapter 2 provided some information to be added to balance the description of different designs (loop type and pool type reactors);
2) reviewed the newly proposed Indian facilities and agreed on their relevance and on the questions to be sent to Indira Gandhi Centre of Atomic Research (IGCAR) for clarification;
3) established tables of facility ranking and priority setting in short, mid and long terms (see Appendix 3). The table on short term may evolve according to the Indian answers.

Furthermore, the participants could agree about the next steps towards the completion of TAREF mandate, including the revision of Chapter 2, the comments on Chapter 3, and the draft conclusions and recommendations expected by 12 February 2010. The draft of the final report is scheduled for 15 March 2010, for submission to the PRG early April 2010.

Summary of actions

First of all, the actions from last meeting were reviewed. Concerning the Russian experimental facilities (action 5), as no appropriate information was received by NEA, they cannot be considered in the TAREF task. Concerning action 10 on possible setting of an experiment to address the safety issue E5 (sodium leak evolution and detection), KIT answered that no funding is presently available.

On the basis of a CSNI recommendation, information on Indian facilities was asked by the NEA Secretariat to IGCAR and was made available for discussion at the meeting.

In the following sections, the actions resulting from the discussions are summarised:

1. The agenda was approved with slight modification in the order of items to fit some needs.
2. The summary of actions of the 3rd TAREF meeting was approved without any changes.
3. The revised Chapter 1 (Introduction) was reviewed and endorsed by the participants provided consideration of two comments from the Chair.
4. Chapter 2 (Outline of SFR Systems) was revised by Hervé Chalaye and Yoshiharu Tobita. The revised version was reviewed during the meeting. Hervé Chalaye will update by 12 February 2010 this chapter according to the participant comments. In particular, he will add the justifications of the new design options of the 1 500 MWe SFR loop reactor, and a footnote for the definition of "practical elimination of reactivity accidents".
5. Some comments on Sections 3.1 and 3.2 of Chapter 3 were given during the meeting. It was agreed to send all the comments to the Chair, with copy to the NEA Secretariat by 12 February 2010.

6. Section 3.3 of Chapter 3 dedicated to "Facilities versus issues" was reviewed in detail. In particular, the availability date of some facilities was checked and if necessary revised. Also, some new facilities proposed by US-NRC were considered in the table.

7. The participants reviewed the updated facility sheets. The sheet on TTS provided by JAEA will be added to Appendix 1 by the NEA Secretariat. The American facilities were reviewed on the basis of the new information provided by US-NRC. For the relevant facilities, Kathy Gibson will distribute by 12 February 2010 facility sheets according to the common format.

8. A plan still exists to construct a new sodium loop in the Nuclear Research Institute Řež in the Czech Republic. Experiences are expected mainly on heat and mass transfer in the cover gas, mechanisms of deposits and the consequences of these deposits, qualification tests of various devices necessary for SFR and the investigations on innovative Energy Conversion System based on Brayton cycle. This information may be confirmed within 2-3 months. Vaclav Dostal will provide information as soon as possible.

9. Hervé Chalaye will prepare by 12 February 2010 a sheet describing ASTRID and the issues that can be addressed by this facility. At a first glance, ASTRID can address B1, B4, B5, C1 (?), F1 and G2. In the long term, ASTRID may be recommended for irradiation purpose and for providing irradiated fuel pins of new designs and at various burn-up levels, to be used for transient tests simulating fuel accidental behaviour to be performed in other facility (for instance in CABRI).

10. The review of the Indian experimental facilities was made with focus on the facility relevance; the conclusions of the group for selection are summarised in Table 4.

11. The questions to IGCAR for additional information will be prepared by the chair and the NEA Secretariat. It was agreed that a separate table will be added in Section 3.3 (facilities versus issues) on the basis of the available information and taking into account the selection made by the group.

12. The group members agreed that for new SFR projects, the most important and top tier R&D safety needs concern the technical areas with the following priority order:

    - Fuel safety (B) and severe accidents (D) issues are of prime interest due to the lack of knowledge on new pin design and materials.
    - Thermal-hydraulic (A) and core physics issues are of second priority as one can live with the current knowledge and considering some margins to cover uncertainties.
    - Sodium risks (E) and structure integrity (F) issues may be considered with third priority as they are more design dependent.

13. On the basis of the above considerations and the table issues versus facilities and their ranking, the participants drew the following preliminary conclusions and recommendations. However, this approach does not cover all the US facilities (see Appendix 4) as the related information was only made available during the meeting. It is to be noticed that the facilities identified for the short-term period (0-3 years) are assumed to be also available for mid (4-8 years) and long terms (>8 years). Facilities available in the medium term are also supposed to be available in long term.

    a. In the short term, data on fuel safety are needed, especially for new fuel pins designs. The Indian FBTR reactor would be a solution in the short term, provided appropriate capability to be confirmed by additional information from IGCAR. Although not dedicated to areas of first priority but owing to their availability and relevance,

KASOLA for thermal-hydraulic issue and SWAT-1R -3R for sodium risks, may be considered in short term.

b. Severe accidents issues can only be addressed from the medium-term period due to lack of available facilities in the short term. IGR which is addressing fresh fuel may be a suitable solution in the medium term as plans exist for it to handle irradiated fuel. VULCANO can also help for severe accident technical area, provided refurbishment for sodium use. TREAT experimental reactor was also considered in medium term, for severe accidents issues but restart of the facility is not yet decided (decision expected in 2010).

c. In the medium term, JOYO was identified as suitable to address fuel safety issues and some others; however, uncertainty still exists as decision for the possible repairing and operating schedule is not yet taken. MASURCA may be suitable for core physics issues.

d. In the long term, ASTRID and JHR may address fuel safety issues.

e. In the long term, the CABRI experimental reactor was recognised by the group members as the most appropriate to address irradiated fuel behaviour under incidental conditions (fuel safety issue) and severe accidents issue.

f. In the long term and for innovative design, LIFUS5 would address sodium interaction with alternative coolant species.

14. Additional relevant US facilities will be included in the tables "facilities versus issues" and, if appropriate, in the tables of "highest ranked facilities" as soon as the related information will be provided.

15. The Chair will complete the draft conclusions and recommendations for distribution by 12 February 2010. Comments by TAREF members are expected by 19 February 2010.

16. The completion of the Draft Report is scheduled for 15 March 2010, for a submission to PRG early April 2010.

Agenda of the fourth meeting

Day 1 ***(Tuesday, 2 February, 2010)***

14:00	Opening and welcome	Chair
14:15	Approval of the agenda	Secretariat
14:20	Approval of the summary record of the 3rd TAREF meeting – Review of actions from the previous meeting	Chair + Secretariat
14:30	Approval of Chapter 1 (Introduction)	All
14:45	Last review and approval of Chapter 2 (outline of SFR systems)	All
15:15	*Coffee break*	All
15:30	Update of the facility sheets	All

16:00	Discussion and approval of Chapter 3 (technical areas, safety issues and facilities): 3.1 Introduction 3.2 Technical areas and related safety issues 3.3 Facilities versus issues	All
18:00	End of day 1	Chair

Day 2 ***(Wednesday, 3 February, 2010)***

09:30	Discussion and approval of Chapter 3 (Cont.)	All
11:00	*Coffee break*	All
11:00	Discussion and approval of Chapter 3 (Cont.)	All
13:00 14:00	*Lunch break*	All
14:00	Discussion of preliminary conclusions and recommendations	All
16:00	Actions for completion of the report	All
16:30	Closure of the meeting	Chair

Appendix 4

LIST OF TAREF MEMBERS

CANADA

Dr. Hussam KHARTABIL	khartabh@aecl.ca

CZECH REPUBLIC

Dr. Vaclav DOSTAL	vaclav.dostal@fs.cvut.cz

FINLAND

Mr. Timo MERISAARI	timo.merisaari@lut.fi

FRANCE

Mr. Pascal ANZIEU	pascal.anzieu@cea.fr
Mr. Daniel BLANC	daniel.blanc@irsn.fr
Mr. Hervé CHALAYE	herve.chalaye@Ms.
Ms. Joelle PAPIN – CHAIR (SFR Systems)	joelle.papin@irsn.fr
Dr. Claude RENAULT	claude.renault@cea.fr
Dr. Daniele VERWAERDE	daniele.verwaerde@edf.fr

GERMANY

Dr. Henrique AUSTREGESILO	henrique.austregesilo@grs.de
Dr. Axel BREEST	axel.breest@grs.de
Dr. Wolfgang HERING	wolfgang.hering@irs.fzk.de
Prof. Dr. Antonio HURTADO	antonio.hurtado@tu-dresden.de
Dr. Th. Walter TROMM	walter.tromm@nuklear.fzk.de

HUNGARY

Mr. Ivan TÓTH	tothi@aeki.kfki.hu

ITALY

Dr. Fosco BIANCHI	fosco.bianchi@enea.it

JAPAN

Mr. Tatsuo IYOKU	iyoku.tatsuo@jaea.go.jp
Mr. Yoshiharu TOBITA	tobita.yoshiharu@jaea.go.jp
Dr. Taisuke YONOMOTO	taisuke.yonomoto@cao.go.jp

KOREA (REPUBLIC OF)

Dr. Yong Wan KIM	ywkim@kaeri.re.kr
Mr. Young-Min KWON	ymkwon@kaeri.re.kr
Dr. Won-Jae LEE	wjlee@kaeri.re.kr
Dr. Joon-Eon YANG	jeyang@kaeri.re.kr

NETHERLANDS

Mr. Michiel HOUKEMA	houkema@nrg.eu

P.R. OF CHINA

Mr. Yujie DONG	dongyj@mail.tsinghua.edu.cn

UNITED STATES OF AMERICA

Dr. Jennifer UHLE – CHAIR (GCR Systems)	jennifer.uhle@nrc.gov
Dr. Said ABDEL-KHALIK	said.abdelkhalik@me.gatech.edu
Ms. Kathy GIBSON	Kathy.Gibson@nrc.org

INTERNATIONAL ORGANISATIONS

European Commission (EC)

Dr. Daniel MAGALLON	daniel.magallon@ec.europa.eu
Dr. Luca AMMIRABILE	luca.ammirabile@ec.europa.eu

OECD Nuclear Energy Agency (NEA)

Dr. Carlo VITANZA	carlo.vitanza@oecd.org
Dr. Abdallah AMRI	abdallah.amri@oecd.org

(The Canadian, Dutch and Korean members did not attend meetings.)

OECD PUBLISHING, 2 rue André-Pascal, 75775 PARIS CEDEX 16
ISBN 978-92-64-99155-2