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EXPERIMENTAL FACILITIES IN SUPPORT OF LIQUID METAL COOLED FAST NEUTRON SYSTEMS

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IAEA NUCLEAR ENERGY SERIES No. NP-T-1.15

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2018

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email: sales.publications@iaea.org
http://www.iaea.org/books

Printed by the IAEA in Austria
October 2018
STI/PUB/1806

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Experimental facilities in support of liquid metal cooled fast neutron systems / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2018. | Series: IAEA nuclear energy series, ISSN 1995–7807 ; no. NP-T-1.15 | Includes bibliographical references.
Identifiers: IAEAL 18-01173 | ISBN 978–92–0–101018–6 (paperback : alk. paper)
Subjects: LCSH: Fast reactors. | Liquid metal fast breeder reactors. | Liquid metal cooled reactors.
Classification: UDC 621.039.526 | STI/PUB/1806

FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

This publication presents both an overview of and detailed information on experimental facilities currently in the design phase, under construction or in operation, in support of the development and deployment of innovative liquid metal cooled (sodium, lead and lead–bismuth eutectic) fast neutron systems, both critical and subcritical. An overview of these facilities is presented in the report, and detailed information is provided in the profiles of each facility on the CD-ROM attached to this publication and available on the IAEA web site. The profiles, providing a thorough technical description of the facilities, including their specific features for utilization, represent the main component of this compendium.

By providing end users with detailed information on existing and planned experimental facilities able to support innovative liquid metal cooled fast neutron systems, this compendium will facilitate cooperation between organizations with an active programme on these nuclear energy systems. As a consequence, it will enhance the utilization of these systems and the involvement of engineers and researchers in the associated experimental programmes for education and training in the field of advanced reactors.

The IAEA wishes to acknowledge the assistance of all the experts who contributed to the development of this publication, in particular M. Tarantino (Italy), O. Gastaldi (France) and A. Sorokin (Russian Federation) for their valuable contributions. The IAEA officers responsible for this publication were M. Khoroshev and S. Monti of the Division of Nuclear Energy and Nuclear Power.

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1. INTRODUCTION

1.1. BACKGROUND

It is generally recognized that long term development of nuclear power as a part of the world's future energy mix will require fast reactor technology with closed fuel cycles [1–4]. Fast reactors can play an important role in meeting growing energy needs worldwide in a safe, environmentally clean and affordable manner due to the unique potential of this reactor technology to become a sustainable energy source. The fast neutron spectrum allows fast reactors to increase the energy yield from natural uranium by a factor of sixty to seventy compared with thermal spectrum reactors. This increased yield would allow for a sustainable fuel supply for nuclear power programmes that would last for thousands of years based on proven uranium resources as well as for a significant improvement of nuclear waste management. It is for these reasons that fast reactors have been under development for decades in several countries, primarily as breeders and, in recent years, also as high level waste burners.

From the very beginning of the development of nuclear energy, the outstanding advantage of liquid metal cooled fast neutron systems (LMFNSs) has been their ability to provide a high breeding ratio — thus allowing the effective use of uranium and thorium resources — as well as the excellent thermal properties of the liquid metal coolants. Additional advantages were the ability of some of them (in particular those with sodium coolant) to tolerate the use of standard structural materials and the possibility of achieving an internal breeding ratio greater than unity (both sodium and lead cooled reactors). The latter feature makes possible core design for long term operation without fuel reloading. Liquid metal coolants are also attractive because they allow a comparatively high coolant temperature to be achieved and therefore allow high thermal efficiency. Currently, several fast reactor systems are being considered, designed and operated worldwide. Among these are sodium cooled fast reactors (SFRs) and heavy liquid metal cooled (HLM) fast reactors, which include (i) lead cooled fast reactors (LFRs), and (ii) reactors cooled by lead–bismuth eutectic (LBE), both critical and subcritical accelerator driven systems (ADSs). Sodium reactors, in particular, have a comprehensive technological basis due to the experimental feedback obtained from the operations of experimental, prototype and commercial sized reactors. The operational experience of HLM reactors is limited to Russian alpha-class submarines. Several countries with nuclear programmes are considering ADSs as a technology to implement nuclear waste transmutation in the scope of their nuclear waste management strategies.

LMFNS is a common term for SFRs and LFRs, including LFR/ADS fast neutron systems (FNSs). The term 'LFR' in this report refers to all HLM FNSs, that is, lead and lead–bismuth cooled reactors, in both critical and subcritical installations.

The status of fast reactor research and technology development is summarized in Ref. [1]. The international experience of SFRs relies on several countries (China, France, Germany, India, Japan, Kazakhstan, the Russian Federation, the United Kingdom and the United States of America) with a cumulative 420 years of operation (as of 2015). Among the fast neutron reactor systems, the SFR has the most comprehensive technological basis thanks to the experience gained from worldwide operation of several experimental, prototype and commercial fast reactors since the 1940s. Reference [1] notes that a "considerable amount of technological experience has already been acquired for the SFR system, thereby providing a sound basis for the further development and eventual deployment of new SFR designs."

Recent developments highlight the observation made in the fast reactors status report: the Russian Federation constructed — and in 2015 connected to the grid — the industrial sized SFR BN-800. Today, there are seven operational reactors, of which five are actually in service: the China Experimental Fast Reactor (CEFR), 65 MW(th), 20 MW(e), in China; the fast breeder test reactor (FBTR), 40 MW(th), 13.2 MW(e), in India; and the BOR-60, BN-600 and BN-800, in the Russian Federation. The Monju and Jōyō reactors in Japan are currently under long term shutdown. Status of Innovative Fast Reactor Designs and Concepts [5] provides a summary of the fast reactors operated so far. The information is also presented in the Advanced Reactors Information System (ARIS) database [6]. Thus, it can be postulated that the SFR system development is based on the vast operational experience of all these reactors.

Furthermore, India is currently constructing a prototype fast breeder reactor (PFBR), 500 MW(e). Several design projects are currently in progress: a demonstration fast reactor, 1000–1200 MW(e), in China; a commercial fast breeder reactor, 600 MW(e), in India; and the sodium cooled fast neutron research reactor MBIR, up to

150 MW(e), in the Russian Federation. The Generation IV International Forum (GIF) is considering six SFR concepts, among them the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID), 600 MW(e), in France; the Japan Sodium Cooled Fast Reactor (JSFR), 750 MW(e), in Japan; the prototype sodium cooled fast reactor (PG-SFR), 150 MW(e), in the Republic of Korea; and the BN-1200, in the Russian Federation. In the USA, licensing activities for a super safe small and simple reactor (4S) designed by the Toshiba Corporation started in 2007 [5]. The HLM reactor was initially developed in the former Soviet Union; its prime objective was the design and construction of nuclear reactors for submarine propulsion (see Ref. [1]). For more than fifty years, major contributions in the development of HLM technology have been made by Russian scientists and industries actively pursuing the usage of lead and LBE as the coolants for nuclear reactor technology. Based on this extensive experience, followed by worldwide research and development (R&D) efforts, several new nuclear reactor designs have been developed for electrical energy production. Currently, several LBE cooled and pure lead cooled reactor projects are under development: CLEAR-I, in China; PEACER-300, in the Republic of Korea; SVBR-100, BREST-300 and BREST-1200, in the Russian Federation; the small, sealed, transportable, autonomous reactor (SSTAR), 10–100 MW(e), in the USA; and MYRRHA, the European lead cooled system (ELSY) and its evolution, the European Lead Fast Reactor (ELFR), as well as the Advanced Lead Fast Reactor European Demonstrator (ALFRED), in the European Union.

The Generation IV (GEN-IV) Technology Roadmap [4] prepared by GIF members has identified six promising advanced reactor systems and related fuel cycles with the potential to meet new technology goals of improving safety, sustainability, economic competitiveness and proliferation resistance. Of these systems, GIF is considering two fast neutron reactors cooled by liquid metal (SFR and LFR).

The SFR system is identified in the GEN-IV roadmap as a promising technology to perform the missions of sustainability, actinide management and electricity production if enhanced economics for the system can be realized [4].

The LFR is identified in the GEN-IV roadmap as a technology with great potential to meet the needs for both remote sites and central power stations [4]. The GEN-IV roadmap also outlined the R&D necessary to develop these concepts for potential deployment.

Committed to sustainability and addressing the progression of technology, the IAEA provides support to Member States covering all technical aspects of current, evolutionary and innovative fast reactor and subcritical hybrid system R&D, design, deployment and operation. Its Technical Working Group on Fast Reactors (TWG-FR) offers the largest international forum for the exchange and transfer of information in the areas of FNSs. The IAEA has been supporting its Member States with a number of activities in the fields of safety, technology development, modelling and simulation, education and training as well as fast reactor knowledge preservation (FRKP).

In the field of LMFNSs the IAEA recognizes the importance of international collaboration in gathering data and developing further knowledge related to FNS development as well as of the verification, validation and qualification (V&V&Q) of simulation codes used for the design and safety analysis of innovative reactors. The IAEA can play a key role in establishing and coordinating international efforts. In addition, international coordination and collaboration allow sharing and the effective use of simulation codes, experimental data and facilities.

The identification of existing experimental infrastructures, as well as of new experimental facilities, based on the recognized R&D needs of Member States with fast reactor programmes is considered a priority. The IAEA actively promotes the harmonization of these efforts at the international level.

Existing and planned experimental facilities in support of the technological development of LMFNSs have been considered at several IAEA technical meetings. To summarize these findings, the LMFNS project has been set up. This compendium has been developed to present data on the facilities in a structured way. This initiative will eventually produce a living database of the same experimental facilities, to be regularly updated by the IAEA and interested Member States.

The choice of coolant has a key influence on reactor options, design rules and safety approach. The coolant in a fast reactor has a lower fraction of the core volume compared with water cooled reactors, but the coolant occupies a significant volume of the fast reactor core. Sodium, lead and LBE are comparatively poor neutron absorbers and moderators and thus are well suited for use as coolants in fast reactors. Many technological problems with sodium coolants have been resolved, but safety remains one of the prime issues of LMFNS development.

A recent survey of the available sodium data literature has indicated the need for consistent and up-to-date sodium property data for use by IAEA Member States. In addition, some Member States have expressed interest in an international effort focused on obtaining and sharing design approaches and guidelines, as well as best practices for the operation and safety of sodium experimental facilities. This knowledge is often experience based and dispersed; it needs to be compiled, structured and disseminated through the IAEA.

For this purpose, the IAEA organized a coordinated research project, Sodium Properties and Safe Operation of Experimental Facilities in Support of the Development and Deployment of Sodium Cooled Fast Reactors (NAPRO). The NAPRO project will support the Member States' SFR research programmes by:

- Collecting, expertly assessing and disseminating consistent sodium property data such as surface tension, saturation vapour pressure and emissivity, thermodynamic behaviour of ternary oxides in sodium and solubility–diffusivity of metallic impurities;
- Developing guidelines and best practices for sodium facility design and operation, including fill and drain, purification, out-gassing prior to filling, sodium storage, component handling and drying sodium piping after repair;
- Developing guidelines and best practices for sodium facility safety, including prevention and mitigation of sodium leaks, prevention and detection of sodium fires, assessment of sodium impact in the environment after accidental release and hydrogen hazards in cleaning facilities.

The NAPRO project is strongly complementary and synergic with the LMFNS compendium project. The NAPRO coordinated research project will greatly benefit from the outcomes of this compendium.

Moreover, the IAEA has been carrying out a dedicated initiative on FRKP since 2002. This initiative was started in response to needs expressed by Member States and within a broader IAEA-wide effort for nuclear knowledge preservation. The goal is to foster the exchange of technical information and to contribute to the preservation of the knowledge base of fast reactor technology.

The main objectives of the FRKP initiative are to:

- Halt the ongoing loss of information related to fast reactors;
- Collect, retrieve, preserve and make accessible existing data and information on fast reactors.

These objectives require the implementation of activities supporting digital document archival, data exchange, search and retrieval. They also require facilitating knowledge preservation over the next decades by developing and using advanced information technology.

For this purpose, the IAEA is developing a new FRKP portal, which provides authorized users with access to:

- Any fast reactor related documentation that Member States wish to share;
- Full papers and other materials from consultancies and technical meetings;
- Documentation from coordinated research projects, including the public and working materials;
- Former reports (which will remain available through the International Nuclear Information System (INIS) repository);
- Document repositories on the shared document server.

Unlike INIS, which manages publicly available information, the FRKP portal is envisioned to control and manage both publicly available and restricted forms of information. The FRKP portal is designed for use by a specific community; however, web pages providing general information about the FRKP portal and taxonomy applied in the portal will be created and published.

The IAEA knowledge preservation initiatives and tools in the field of FNSs are designed to be of interest to national nuclear authorities, regulators, scientific and research organizations, commercial companies and all other stakeholders involved in fast reactor activities at the national or international level. This includes designers, manufacturers, vendors, research institutions, academics, technical support organizations and safety authorities.

1.2. OBJECTIVE

The objective of this publication is to create a comprehensive compendium providing detailed information on experimental facilities currently being designed, under construction or operating, in support of the development and deployment of innovative LMFNSs (sodium, lead and LBE), both critical and subcritical.

This compendium aims to enhance the use of facilities by providing end users with detailed information. It will identify existing or future operational experimental facilities able to support innovative LMFNSs, and encourage international collaborations.

1.3. SCOPE

The scope of this compendium covers two main types of experimental facilities:

— Those devoted to SFR development;
— Those devoted to LFR system development.

For each facility presented in this compendium, the following data are outlined:

— General information;
— Status of the facility;
— Main research field(s);
— Technical description;
— Completed experimental campaigns: main results and achievements;
— Planned experiments;
— Training activities;
— References.

In total, this compendium and the related database [7] cover information on about 150 LMFNS experimental facilities. There are approximately the same number of facilities in support of SFR and LFR development and deployment: 79 for SFRs and 72 for LFRs. In addition to the facilities in support of SFRs and LFRs, there are a few dual application facilities in support of both designs. These cross-cutting facilities are noted in the overview tables in Section 4 of this report.

General purpose facilities and laboratories (i.e. creep laboratories, metallographic laboratories, etc.) are not included in this compendium as they are considered beyond its scope. Research reactors are also considered beyond its scope and are included in other compendiums. Facilities devoted to the development of high power proton accelerators for ADS are not considered in this compendium. Facilities planned to be operated beyond 2020, even if under design or construction, are considered beyond the scope of this compendium. Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. STRUCTURE

Section 2 presents an overview of international experience in the development of SFRs, and extensive R&D efforts ongoing worldwide in the area of LFRs.

In Section 3, an overview of experimental activities in support of LMFNSs and an outlook of experimental studies at existing facilities are presented.

An overview of experimental facilities currently being designed, under construction or operating, in support of the development and deployment of LMFNSs, is presented in Section 4. The facilities are structured by their applications, namely, by the capabilities and capacities for R&D in certain research areas: zero power facility for V&V&Q and licensing purposes; design basis accidents (DBAs) and design extension conditions (DECs); thermohydraulics; coolant chemistry; materials; systems and components; instrumentation; and in-service inspection and repair (ISI&R). Section 4 of this report guides a user through the individual profiles that provide technical descriptions of the facilities in support of the development and deployment of innovative LMFNSs, including their specific features for utilization. The profiles describe LMFNS facilities using a template agreed to by the broad group of international experts who contributed to this report. Most papers have been provided to the IAEA Secretariat by institutions in IAEA Member States in response to requests for input to this. This compendium also includes some papers that do not follow the required template and format exactly.

The profiles are arranged in alphabetical order by country, in two folders on the attached CD-ROM and in related media storage: The first folder contains the profiles of facilities in support of FNSs, based on SFRs. The second folder contains those in support of LFRs.

An outlook on future development of the facilities, including best practices on desirable construction of new facilities, is presented in Section 5. Ideas and suggestions on using this compendium to establish cooperation between organizations with active programmes on LMFNSs are presented in Section 6.

1.5. USERS

This compendium will be useful for a wide range of governmental and private sector organizations responsible for the development and/or deployment of innovative FNSs in countries with active programmes on these nuclear energy systems, including designers, manufacturers, vendors, research institutions, academics, technical support organizations and other organizations directly involved in technology development programmes on FNSs and, more generally, on advanced nuclear energy systems.

It is expected that the detailed information on existing and future experimental facilities able to support innovative LMFNSs provided in this compendium will facilitate cooperation among organizations. Consequently, it is expected to enhance the use of those facilities as well as to involve engineers and researchers in the associated experimental programmes for education and training in the field of advanced reactors.

2. FAST NEUTRON SYSTEMS: OVERVIEW OF STATE OF THE ART, DESIGN OBJECTIVES, DEVELOPMENT TARGETS AND MAJOR R&D TOPICS

2.1. SODIUM COOLED FAST REACTORS

2.1.1. State of the art and international experience on SFRs

The international experience of SFRs, as presented in Table 1, relies on several countries (China, France, Germany, India, Japan, Kazakhstan, Russian Federation, UK, USA) with 420 cumulative years of operations (as of 2015). Today, there are seven operational fast reactors, of which five are in service (BOR-60, BN-600, BN-800, FBTR, CEFR) and two are under long term shutdown (Monju and Jōyō in Japan). One more reactor is under commission (PFBR in India). Thus, the SFR systems have vast operational experience associated with all these reactors.

A detailed technical analysis of this experience is the subject of several documents (e.g. see Ref. [8]). A brief summary that highlights the achievements and intrinsic advantages of the SFR system is provided.

The operation of SFRs has demonstrated the excellent use of uranium resources as well as the capability of these reactors to recycle the plutonium without any limitation in the number of recycling operations (multi-recycling). Unlike the vast majority of the reactors currently operating or under construction all over the world — which consume less than 1% of natural uranium to extract the energy contained in it — SFRs have the capability to consume, in theory, almost the whole resource via multi-recycling of the successive used fuels.

The primary system is operated at low pressure (not pressurized) but it has a very high thermal inertia, which provides operators with significant time to intervene in case of loss of cooling. In operation, there is a high margin

TABLE 1. INTERNATIONAL EXPERIENCE WITH SFRs *(as of 2015)*

Reactor (country)	Thermal power (MW)	Startup	Final shutdown	Operation lifetime (years)
EBR-I (USA)	1.4	1951	1963	12
BR-5/BR10 (Russian Federation)	8	1958	2002	44
DFR (UK)	60	1959	1977	18
EBR-II (USA)	62.5	1961	1994	33
Fermi 1 (USA)	200	1963	1972	9
Rapsodie (France)	40	1967	1983	16
SEFOR (USA)	20	1969	1972	3
BN-350 (Kazakhstan)	750	1972	1999	27
Phénix (France)	563	1973	2009	36
PFR (UK)	650	1974	1994	20
KNK-II (Germany)	58	1977	1991	14
FFTF (USA)	400	1980	1993	13
Superphénix (France)	3000	1985	1997	12
Jōyō (Japan)	50–75/100/140	1977	Operational	34
Monju (Japan)	714	1994	Operational	17
BOR-60 (Russian Federation)	55	1968	Operational	46
BN-600 (Russian Federation)	1470	1980	Operational	34
FBTR (India)	40	1985	Operational	28
CEFR (China)	65	2010[a]	Operational	4
BN-800 (Russian Federation)	2100	2014[b]	Operational	
PFBR (India)	1250		Under commission	
		Total		420

Notes: [a] CEFR commissioning tests were performed at different power levels in December 2014.
[b] First criticality of BN-800 occurred on 24 June 2014. Connection to the grid took place in December 2015.

with the sodium boiling temperature, typically 300°C. The oxide fuel is more mature when compared with the limited experience feedback on dense fuels (carbide, nitride, metal).

Controlling the reactor appears easy owing to the:

— Absence of burnable poisons (to compensate for the excess reactivity), in contrast to pressurized water reactors (PWRs);
— Absence of poisoning effect generated by highly neutron absorbing fission products such as xenon or samarium in PWRs;
— Self-stabilizing thermal feedback.

Experimental studies and the operational experience of active and passive decay heat removal (DHR) systems, based on two types of cold sources (air and water), have demonstrated their efficiency. GEN-IV reactors will aim for higher capabilities of these systems to further improve the safety of these facilities.

The environmental assessment of these facilities is very positive and the collective dose workers are exposed to is very low compared with that from other types of reactors.

The usage of sodium also presents some issues requiring special attention and design solutions. The main drawback of liquid sodium is its chemical reactivity with oxygen (combustion) and with water, which can result in fire or explosion. The main responses to these potential reactions consist in improving the tight and resistant barriers to avoid contact between sodium and these compounds and to limit the consequences of the potential occurrence of fires or explosions. The second drawback is related to the positive void coefficient (particularly for the large cores). This risk can be minimized by optimizing the core design to limit the probability of events leading to local sodium voiding. In addition, sodium is opaque and the temperature to be considered during shutdown is relatively high (180–200°C), which makes inspections more complex. The issues above are well known from numerous past studies in which efficient countermeasures have been developed to reach the safety goals requested for next generation nuclear power plants (NPPs).

For the future, several new innovative designs are planned with sodium cooled power reactors (CEFR-600 in China, ASTRID in France, FBR 1 and 2 in India, JSFR in Japan, PG-SFR in the Republic of Korea, BN-1200 and MBIR (up to 150 MW(th)) in the Russian Federation), as well as small modular reactors using sodium as a coolant (e.g. 4S, Toshiba/Terrapower). Given the importance of the existing experience and taking into account new regulatory requirements aimed at meeting the highest level of safety, further needs to improve fast reactor technology are emerging.

2.1.2. SFR design objectives

As already pointed out, the SFR has the most comprehensive technological basis of any FNS as a result of the experience gained from worldwide operation of several experimental, prototype and commercial size reactors. It is therefore possible to begin designing and constructing a prototype of a demonstrator commercial SFR.

The following information was published in Ref. [9].

Currently, the design of SFRs worldwide aims to achieve general goals [10], which can be summarized as follows:

— Demonstration of a large scale power plant connected to the grid to guarantee the representativeness of the reactor core and main components. Such a facility could compensate for the operational costs by generating a significant amount of electricity.
— Design flexible enough to be able to eventually test innovative options that have not been chosen for the initial design. Testing of novel instrumentation technologies, new fuels and even new system components should be manageable in the new generation demonstrator commercial SFR.
— Commissioning with approximately the same time frame as Generation III power plants.
— Safety levels of SFRs at least equivalent to those of GEN-III reactors, taking into account the lessons from the Fukushima accident. The focus will be on validating safety measures, enabling future reactors to ensure even more robust safety compared with GEN-III reactors (e.g. in case of a core meltdown accident categorized as a DEC).
— Higher level of availability and higher loading factors can be reached.

— Availability of the demonstrator SFR for irradiation experiments without being a material testing reactor. These experiments will help improve the performance of the core and absorbers, as well as test new fuels and structural materials such as carbide fuel and oxide dispersion steel (ODS) cladding. For examining irradiation objects, the SFR demonstrator should be equipped with a hot cell located either in the plant or nearby.
— Demonstration of the viability of transmuting radioactive waste to reduce the volume and long term radiotoxicity of final radioactive waste.
— Integration of feedback from past reactors, while being clearly improved and belonging to GEN-IV. Therefore, the design will take into account current safety requirements, especially in terms of protecting against both internal and external acts of malevolence, as well as protecting nuclear materials. It will also meet the latest requirements in terms of proliferation resistance, and maintain its costs by following a value analysis approach starting from the design phase.

Concerning the design of future SFRs [11], pool type and loop type reactors are considered, with associated options for heat transfer (pumps and intermediate heat exchangers (IHXs)) from the core to a secondary sodium circuit. Several options for a fuel handling system are considered such as whole core discharge or not, and fuel assemblies with or without minor actinides.

The energy conversion system (ECS) in the mainstream design concepts is based on the Rankine cycle (water/steam): primary sodium and water/steam are separated by an intermediate circuit, filled with sodium coolant. The main components of this circuit are an IHX between the primary sodium and intermediate sodium, a steam generator unit between the intermediate sodium and water/steam, and a secondary pump. The overall net targeted efficiency could be above 40%.

An alternative option is a Brayton cycle [12], which avoids a sodium–water reaction if steam generator tubes rupture. Different options for the fluid choice are currently being studied (e.g. nitrogen, supercritical CO_2). The target in terms of efficiency could be somewhat lower or higher than that of the Rankine cycle, depending on the choice of fluid, but it requires specific technical development and qualification of the main components of this cycle with sodium technology (i.e. heat exchanger, turbo machinery, etc.).

2.1.3. Experimental activities and major R&D topics in support of SFR development

The experimental activities in support of SFR development respond to diversified needs, in terms of topic, but also in terms of the nature of the tests. Indeed, two kinds of tests had to be covered:

— Fundamental physical tests;
— Qualification of components, systems or circuits at a scale representing the components of the reactor.

The experience with SFRs is very important. The associated experimental facilities in support of SFR development have been permitted to run many different experimental activities. However, many more experimental activities are still desired to respond to emerging needs driven by new objectives of SFR development which are, in turn, driven by new requirements. Indeed, most SFR projects aim to respond to the targets defined by the GIF. Innovations in the development of new SFRs and support of existing ones are the major motivations for experimental activities.

Some, but not all, of these experimental needs require the presence of sodium. In some cases, representativeness can be achieved by using simulant fluids (water, air, sodium–potassium alloy). In this compendium, experimental facilities that can use water and air as simulant coolants are also presented.

Consequently, the main R&D requiring experimental activities is driven by the following major topics considered in innovative SFR designs.

2.1.3.1. Thermohydraulic behaviour for operation and safety

This large topic covers many subjects. Of course, it relies on past experience and several specific codes, but some complementary experimental validation and qualification remain necessary, such as:

— Thermohydraulics of the primary circuit (for new reactor designs), the natural convection of the primary loop (in particular, for the feasibility demonstration of a passive DHR system);
— Behaviour of subassemblies as well as studies associated with potential sodium boiling situations;
— Gas entrainment and creation of a vortex on free surfaces;
— Evaluation of sodium aerosol behaviour on the gas plenum and its influence on the heat transfer coefficients, upper core structure vibrations, and thermal fatigue of the upper core structure due to core outlet temperature fluctuations;
— Thermohydraulics of main components such as IHX or steam generator;
— The internal thermohydraulics of the fuel subassembly (pressure drop, cavitation, effects of vibration, etc.);
— Thermohydraulics of gas when using the Brayton cycle for the power conversion system;
— Thermal stratification and mixing phenomena in the primary vessel (fluctuations of the temperature at the boundary of a stratified region).

Some of these tests can be performed using water. That is why some water facilities are listed and detailed in the section on facilities in support of SFRs. These facilities are devoted to covering basic thermohydraulic issues.

However, in some cases, mainly when a high temperature range of tests of the proper heat exchange conditions are required (such as for the behaviour of sodium aerosols, final qualification of subassemblies or operation of a new steam generator design), some sodium facilities have to be used. It can be a relatively large sodium facility, implying an important investment cost.

2.1.3.2. Improvement of system reliability and operation: Availability, safety and investment protection

This objective relies on the performance of instrumentation for continuous monitoring as well as on ISI&R. Several needs have been underlined to cover this topic, which depend on reactor design choices. Some of them requiring R&D and some examples of sodium experiments are discussed further on.

For continuous monitoring, further development of techniques and/or performing experiments of eddy current flowmeter for sodium flow measurements at the core outlet are needed. The examples include:

— Development of high temperature ionization chambers;
— Measurements of concentrations of oxygen and hydrogen in sodium;
— Techniques for characterization of gas content;
— Detection of sodium–water reaction;
— Fuel assembly identification;
— Sodium free surface level measurements;
— Core geometry measurements;
— Sodium leak detection.

For ISI&R, the following are needed:

— Ultrasonic under-sodium viewing (global position checking);
— Ultrasound of sodium telemetry and surface metrology (accurate location checking);
— Ultrasonic non-destructive examination (cracks, corrosion or thinning detection) applied to different components and systems (main welding in primary vessel, sodium gas compact heat exchangers, etc.).

In complement to the development of sensors and signal treatment, this domain could require carriers and robotic arms and lead to a dedicated R&D programme requiring analytical sources to validate elementary choices as well as large sodium experimental pots (a few metres in diameter) in order to qualify the developed technical solutions. Another field of activity is the development of under-sodium repair techniques.

For all these kinds of development, specific experimental facilities are required, using sodium, in most cases, as a coolant. However, small water installations can be used to develop acoustic techniques in representative conditions. To cover this need, many facilities have been identified. In general, these facilities are not very big but are versatile in their application.

2.1.3.3. *Improvement of decay heat removal for safety*

DHR is a major challenge for all types of nuclear reactors. For SFRs, passive DHR based on sodium natural convection could be a decisive argument in favour of this kind of fast neutron reactor. The behaviour of these systems operating in natural convection is key to demonstrating its reliability in case, for example, of total plant black out.

System analysis, thermohydraulics and computational fluid dynamic codes are the key tools for system calculations and simulations. A qualification study of these systems and validation of these codes have to be carried out based on some experimental validation. This can require a medium scale facility with a devoted design to confirm the performance of the components involved in such a loop but also the operation of a complete loop under natural convection. Several countries are currently working on developing such facilities.

2.1.3.4. *Improvement of reactivity control for safety*

In order to improve the reactivity control of SFRs, further optimization of its core design is required. However, improvement of the control rods and shutdown system are also needed to implement the passive safety systems' deterministic safety approaches. For example, hydraulically suspended control rods or some other technical solution, such as a high temperature trip threshold, could be used based on different physical principles for a system shutdown. Consequently, qualification of shutdown systems in representative conditions (hydraulic tests to study vibrations, risks of uploading, pressure drop, cavitation, etc.) and mechanical tests to demonstrate the operational ability of the shutdown system in relevant normal or abnormal conditions are needed.

These needs require a relatively large sodium facility with a large sodium flow at high temperature (up to around 700°C).

2.1.3.5. *Optimization of the handling route: Availability and economics*

Handling the fuel subassemblies in an SFR is significantly different from handling rod bundles in water reactors. First of all, the opacity of sodium requires working 'blind' as long as the fuel subassemblies are inside the reactor or in the sodium storage tank. Systems to check for movements and obstacles have been developed to remedy this drawback — ultrasonic 'viewing', in particular. Then, the subassemblies need to be cleaned of the sodium which may remain attached to them before they can be stored in water. These operations require radiological protection and are performed using remote controlled equipment.

The experience feedback showed a gradual extension of the duration of the core renewal campaigns, due, on the one hand, to equipment ageing (more frequent failures) and, on the other hand, to stricter assembly movement control procedures requiring a greater number of checks and hold points during operation. Additional R&D is necessary to improve the handling and cleaning speeds to preserve optimum reactor availability.

The reliability of this fuel handling route is becoming a critical issue for the reactor concept without an external storage vessel. Two main constraints have to be considered: the handling of assemblies with high residual power and the requirement to treat fuel assemblies on-line — from the sodium internal storage to the used fuel assemblies' pool.

These constraints induce the necessity to develop more innovative handling systems, with more efficient fuel assembly cleaning processes (to be defined and qualified).

To validate the fuel handling systems, the following R&D needs have been identified:

— In-air tests for handling mechanical devices such as fixed arm charge machines and direct lift charge machines;
— Tests in sodium to examine the behaviour of some handling systems in argon and sodium aerosols, as well as the seismic behaviour of some handling systems.

2.1.3.6. *Design simplification: Economics, performances and periodical inspection*

This topic covers very different actions. It can concern the primary vessel and its internal design (e.g. to be able to address all the periodic inspections), but also the development of electromagnetic pumps for the secondary circuit (components requiring few maintenance actions) or the arrangement of inlet and outlet piping of the main components of the secondary sodium loop. For a large electromagnetic pump, for example, the development of tools that allow confirmation of the design represents a potential field of R&D. In support, experiments using sodium are needed. This kind of R&D study requires a sodium facility with sodium flowing at high speed (10 m/s^{-1}) and devoted instrumentation to quantify magneto-hydrodynamic instabilities.

2.1.3.7. Improvement of core performances: Availability and economics

This topic requires the technological programme already mentioned (see Section 2.1.3.1), as well as neutronics calculations and potential support of some zero power facilities. The development, validation and qualification of coupled codes (thermohydraulics and neutronics) are therefore key issues.

2.1.3.8. Improvement of containment control, for safety, including sodium risks

Considerations for the improvement of containment control include the following:

— The behaviour of sodium aerosols in the atmosphere following a large sodium fire could require some R&D support. Indeed, such studies could expand the understanding of processes within the safety demonstration, e.g. by developing a better knowledge of sodium carbonization in contact with air.
— Depending on the insulating material choice, the issue of corrosion in case of a sodium leak can be a concern.

2.1.3.9. Elimination of occurrence of a large sodium water reaction: Economics, availability and safety

The risk of sodium–water interaction is a major concern when using the Rankine (water steam) cycle. Sodium in the secondary circuit and water in the tertiary circuit are respectively circulating outside and inside the steel tubes of a steam generator. If a tube ruptures, the resulting interaction can be accompanied by relatively complex phenomena (such as wastage and multiple tube ruptures).

Past experimental programmes provided substantial results concerning leak flow rate evolution, pressure wave propagation and mass transfer within the secondary circuit, damage caused to the adjacent exchange tubes and problems (efficiency and rapidity) arising from the sodium–water reaction detection. Representative tests can be envisaged in order to complete an experimental database using different codes to represent these phenomena.

A sodium–water–air reaction occurs when there are two leaks (water and sodium pipes or envelopes) on the same premises due to an external accidental event that ruptures both the sodium and water circuits of the steam generator. The risk of explosion for hydrogen in the presence of air has to be considered. Development of the explosion process involves complex phenomena and their interactions, such as:

— Pressure peaks;
— Gaseous bubble growth;
— Combustion, explosion;
— Sodium fragmentation.

There is a need for a validated model for these phenomena.

Another way to drastically reduce the risk of sodium–water reaction could be the use of another ECS, such as a Brayton cycle, with pressurized nitrogen or supercritical carbon dioxide as a coolant.

An important field of R&D concerns the components of this innovative tertiary loop: primarily, the heat exchanger and turbine. For the heat exchanger, a compact design using printed circuit heat exchanger type modules to get higher heat power density compared with a standard shell and tubes heat exchanger type could be preferable.

However, a printed circuit type heat exchanger has never been used with these fluids industrially and it is, therefore, necessary to:

— Test mock-ups to qualify the concept;
— Quantify the heat exchange correlations and investigate the spatial distribution of fluid flows;

— Investigate challenges of using sodium in narrow channels (draining, cleaning, potential self-plugging, stop/restart, inspection, etc.).

In response to various requirements, for example, concerning supercritical CO_2, studies are dedicated to the:

— Stability of operational conditions near the critical point;
— Industrial development of the turbine;
— Na–CO_2 interaction, which has fewer potential consequences than Na–H_2O interaction, particularly with regard to the wastage effect.

The whole operation of a gas circuit cycle should be evaluated for all conditions (normal and accidental) with support of modelling a system loop, eventually.

2.1.3.10. Cross-cutting topics: Material studies and improvement of system reliability

The SFR system raises a number of material issues, such as corrosion phenomena, due its environment. The following corrosion effects need to be studied:

— Stainless steel in contact with high quality sodium (low impurities level — lower oxygen concentration);
— Related mass transfer, mechanical behaviour of structures for vessels, pipes and internal components;
— Special focus on cladding material used for the fuel assemblies.

The main goal is to confirm the performance of new structural materials, for example, for cladding (AIM2, ODS):

— New hard coating materials (tribology studies are then required) with regard to the expected operating conditions (high burnup, temperature, dose, stress, etc.);
— New, innovative ECS, etc.

2.1.3.11. Improvement of behaviour in severe accident conditions

The development and qualification of severe accident codes and mitigation devices for SFRs require a comprehensive experimental programme that can encompass:

— In-pile experiments;
— Prototypical corium experiments;
— Simulant material tests.

Depending on the pin design, in-pile experiments, in particular, could be necessary to study the behaviour of pins during severe accident transients and to test in-core mitigation devices.

In order to better understand the phenomena during severe accidents and to qualify technical solutions to mitigate their consequences, corium experiments are required, first on a small and medium scale, then at a larger scale, mainly for:

— Fluid-corium interaction;
— Corium relocation;
— Core catcher issues.

2.2. HEAVY LIQUID METAL COOLED FAST NEUTRON SYSTEMS

2.2.1. State of the art and international experience on LFRs

As noted in Section 1.4, it is assumed that the term LFR includes all HLM FNSs, that is, lead and LBE cooled reactors, both critical and subcritical reactor systems.

Research and design on the use of the LBE alloy as a coolant for nuclear reactors was initiated in the early 1950s in the Soviet Union for military submarine propulsion. The first nuclear submarine with an LBE cooled reactor was put into operation in 1963. In total, 15 reactors have been built, including three land system reactors, plus one replacement reactor for submarines. Civil LBE cooled and lead cooled reactor projects have been developed in the past 20 years, among them CLEAR-I, in China; several projects in Japan; PEACER-300 in the Republic of Korea; BREST-300 and BREST-1200 in the Russian Federation; SSTAR in the USA; and MYRRHA, ELSY and its evolution ELFR as well as the LFR-Demo ALFRED within the frame of European projects. Meanwhile, the MYRRHA project has obtained financing from the Belgian government to prepare the front end engineering design, a big step forwards in the realization of an HLM based technology pilot plant in Europe. In the Russian Federation, the BREST-300 design has been completed and is currently prepared for licensing.

2.2.2. LFR design objectives and ongoing R&D

Extensive R&D efforts are ongoing worldwide, addressing issues related to lead and LBE technology concepts, including those performed under GIF. Research activities are ongoing and are expected to continue in the future, with the target to design and construct an LFR/ADS demonstrator.

R&D efforts are necessary to complete the design and support the pre-licensing; this starts with the construction of such systems. Such activities require identifying the technological gaps, which, in some cases, are design dependent and represent the key topics for LFR/ADS developments.

The technological issues of LFR development can be summarized by the following main topics [13–15].

2.2.2.1. Material studies and coolant physical chemistry: Operation

Corrosion of structural materials in lead alloys is the main issue for the design of LFR/ADS. The topic is related to lifetime limits and circuit design. The following issues relate to the corrosion/dissolution in lead alloys:

- Fuel cladding;
- Vessel;
- Reactor internals, components and heat exchangers;
- Corrosion at high temperatures (related to development for long term perspectives);
- Corrosion/erosion of pump impeller materials;
- Corrosion inhibitors development (i.e. coating).

Efforts have been devoted to short/medium term corrosion experiments in stagnant and flowing LBE. Few experiments have been carried out in pure lead and new experimental campaigns are planned. Research activities are still needed on medium/long term corrosion behaviour in flowing lead and LBE under oxygen control.

Tests at higher temperatures (at least 650°C) and greater velocities (up to 2 m/s) are still required. They have been identified to cover long term development of the technology as well as to perform specific tests under particular conditions (i.e. DECs).

The embrittlement and degradation of structures by liquid metal is still an open issue. The condition represents the lower bound of mechanical properties of materials exposed to liquid metal. It consists of the reduction in fracture toughness and of ductility of unirradiated materials after long term exposure to lead and lead alloys. In order to be reproduced with a typical standard procedure, it is necessary to experience and standardize tests with respect to:

- Liquid metal embrittlement (LME);
- Fatigue;

— Creep;
— Stress corrosion cracking;
— Fretting in HLM.

These experiments are necessary and should be conducted for both static and flowing conditions in testing machines where the specimens are exposed to the liquid metal.

Full development of GEN-IV programmes foresees the increase of reactors' operating temperatures (beyond 550°C). This evolution is aimed at enhancing thermodynamic efficiency and introducing cogeneration processes. This challenging goal requires testing:

— 'New' materials such as ODS steels, refractory alloys, SiC composites, MAX phase materials;
— Coated materials.

Moreover, the reactor vessel, the structural materials, the internals and the fuel cladding are subjected (to different extents) to several degradation mechanisms such as neutron irradiation, thermal ageing and corrosion.

In LFR systems, research activities generally aim at understanding, quantifying and predicting such effects on critical components of an NPP. Focus is given to the:

— Performance of the materials in terms of neutron irradiation induced embrittlement;
— Behaviour of stress corrosion;
— Neutron irradiation induced effects such as creep and swelling.

The main objective is to determine whether irradiation will promote embrittlement and corrosion attack by HLM.

The current status of knowledge is not completely addressed, and more experimental investigations are needed to provide high quality data on material behaviour. Assessments of fuel cladding and structural core materials, subjected to both high temperature in a lead environment and fast neutron flux, are critical.

The following issues on irradiation performance of candidate materials are of primary importance for LFR system development:

— Corrosion in HLM under irradiation (coated and uncoated material);
— Irradiation embrittlement of selected materials;
— Irradiation creep;
— Swelling.

The impact of irradiation on materials is a critical issue for LFR development. The experimental infrastructures needed to address these issues are the currently available irradiation machines and research reactors.

The relatively high velocity of liquid metal implies that structural materials are possibly subject to severe corrosion/erosion conditions that might not be sustained in the long term. The materials of the pump impeller have to satisfy demanding requirements which deserve specific experimental installation, such as:

— Capability to withstand exposure to high temperature lead (up to 480°C usually, and higher for long term perspective);
— Capability to withstand corrosion/erosion effects due to high relative coolant velocity (10 m/s, and up to 20 m/s);
— Demonstration of reliability and performances of the pump for a long term operation.

Chemistry control of the coolant and cover gas is another critical issue when operating LFRs. It is essential to control the concentrations of impurities, because of the potential for activation and also because of the possible effect on corrosion, mass transfer and scale formation at heat transfer surfaces. Therefore, the coolant chemistry control includes:

— Oxygen control;

— Impurity source term studies;
— Mass transport;
— Filtering and capturing techniques.

The following specific issues are considered:

— Coolant quality control and purification during operation (i.e. oxygen control — in particular, its thermodynamic control oxygen sensor reliability — coolant filtering, HLM purification and HLM cleaning from components);
— Cover gas control (i.e. radiotoxicity assessment of different elements, migration flow path into cover gas, removal and gettering).

2.2.2.2. Core integrity, moving mechanisms, maintenance, ISI&R: Operation, safety and physical inspection

The fuel manipulator (or handling) system is used to control the reactor power distribution (by off-line shuffling). The system is also used to store and handle fuel assemblies during its overall lifetime (i.e. from the arrival up to spent fuel storage).

In current LFR/ADS designs, refuelling and shuffling are performed remotely. The design and operation of such a machine has to be tested before installation in the reactor, to demonstrate its capability to fulfil its functions reliably and safely. This requires the assessment in an experimental facility of the prototype machine as well as of its components, for qualification purposes, using the following sample experimental activities:

— Cold testing of components (in-air);
— Testing of submerged components;
— Testing the reliability of fuel handler components;
— Integral tests.

In this frame, large pool facilities (i.e. CIRCE, COMPLOT, CLEAR-S — profiles are presented on the CD-ROM accompanying this compendium) might be, in principle, suitable for addressing the issues, owing to their high flexibility. The fuel assemblies maintain and position the fuel within the reactor core, cool the fuel and ensure shielding from radiation streaming from the core. Therefore, different issues need to be tested to verify the suitability of the design features, including ensuring the structural integrity of the fuel assembly. In particular, the following issues need to be considered:

— Mechanical and structural integrity of the fuel assembly, in connection with the fuel loading procedure;
— Wide range of operating conditions;
— Flow induced vibrations;
— Spacer grids — fuel pin interactions.

Moreover, the impact of neutronics on control and shutdown systems should be taken into account for in-core design. The shutdown systems should be designed to meet requirements for the normal and abnormal, up to accident conditions.

The following areas linked to the core's cooling ability need to be investigated:

— Integrity and behaviour of the safety systems (i.e. control rods);
— Loss of core cooling ability and integrity;
— Identification of the initiating events, including those connected with the control rod mechanism design (e.g. control rod withdrawal and ejection);
— Fuel coolant interaction;
— Fuel degradation mechanisms and behaviour (up to its release in the primary system);
— Dispersion and relocation of fuel in the primary system;
— Control rod design and the mechanisms themselves;
— Testing of control rod mechanism operation and performance;

— Reliability of control rod mechanisms and components;
— Impact of seismic loads and sloshing: demonstration that the system is qualified by design to withstand a basis earthquake, i.e. permitting shutdown rods to drop into the core.

2.2.2.3. Steam generator functionality and safety experimental studies: Operation and safety

LFRs are pool type reactors, which have the steam generators (or the heat exchangers) inside the reactor vessel. This implies that an interaction between the secondary circuit coolant and the HLM may occur. Thus, the primary to secondary leak (e.g. steam generator tube rupture, or SGTR) is considered a safety issue in the design and also in the preliminary safety analysis of these reactor types.

There are two major topics of investigation in an SGTR postulated event:

(1) Understanding the phenomena involved in the accident scenario;
(2) Studying how to prevent or mitigate the consequences of an event, reducing the primary system pressurization.

The areas of investigation include:

— Pressure wave propagation across the primary system;
— Sloshing;
— Steam transport in the primary system;
— Steam entrainment into the core;
— Lead/LBE–water interface phenomena;
— Rupture/leakage detection systems;
— Tube rupture countermeasures.

Concerning the steam generator, the main qualification studies regard the:

— Design validation;
— Unit isolation on demand;
— Pressure drop characteristics;
— Behaviour of the components in normal operation (e.g. forced, mixed and natural convection), operational transients and accident conditions.

Apart from the steam generator, there are several auxiliary systems in a reactor which need to be qualified for nuclear applications, such as:

— DHR system;
— Dip-coolers and isolation condenser;
— Reactor vessel auxiliary cooling system;
— Fuel assembly transport system;
— Spent fuel assembly transport and cooling system.

2.2.2.4. Thermohydraulics: Operation and safety

For several years, many countries have conducted research on HLM thermohydraulics. Nevertheless, open issues are pending for LFR development.

The objectives of the activities in relation to pool thermohydraulics are twofold:

(1) To gather experimental data in geometry and with boundary conditions which may improve the knowledge of phenomena/processes at component and system levels;
(2) To generate databases to support the development of and demonstrate the capability of computer codes to predict phenomena/processes relevant to design and safety.

The following topics are identified as relevant at the component and system levels:

- Flow patterns in forced convection, including:
  - Mixing;
  - Stratification (inducing thermal stresses);
  - Stagnant zones;
  - Surface level oscillations.
- Transition to buoyancy driven flow.
- Natural convection flow:
  - Pressure drop;
  - Surface level oscillations.
- Fluid-structure interaction.
- Thermal fatigue issue.
- Sloshing due to seismic event tests.

Moreover, it is important to test the fuel assemblies based on thermohydraulics parameters (i.e. pressure losses, flow distribution, velocity field, clad wall temperature distribution, etc.) and geometrical features, such as rod bundle lattice, subchannel geometry and spacer grids.

The following partial list of topics should be investigated to support LFR system development:

- Heat transfer in forced and natural convection (including transition);
- Subchannel flow distribution;
- Subassembly flow distribution (i.e. open wrap and wrapless);
- Cladding temperature distribution and hot spots;
- Pressure drop;
- Fluid–structure interaction;
- Flow induced vibrations;
- Grid-to-rod fretting;
- Fuel assembly bow.

The topics of investigations also involve the study of the potential sources of core damage and the consequences of such accidents.

Integral tests are mandatory in this frame since full-scale facilities are not available. Data can be extrapolated to the full scale, if the test facilities and the initial and boundary conditions of experiments are properly scaled (i.e. the scaling will not affect the evolution of physical processes important for the postulated accident scenario). This evaluation determines whether the data may be used in nuclear plant safety analyses.

On the other side, integral tests are fundamental for supporting development and demonstrating the reliability of computer codes in simulating the behaviour of an LFR during a postulated accident scenario; in general, this is a regulatory requirement. Applications of computer codes to accident analyses require the implicit assumptions that these codes have the capability to scale up phenomena and processes from test facilities to full-scale plant conditions. However, the different geometric scale characterizing any facility does not ensure a priori that a code, which is able to reproduce a generic transient in a scaled facility, is also able to simulate the same transient in a full scale LFR with the same accuracy.

These considerations imply that integral tests are unavoidable, and complex activities are needed which involve the following objectives and areas of investigation:

- Phenomena and processes at system level connected with design, safety and operation issues;
- Flow blockage studies and related experimental/modelling investigations;
- Simulations and analyses of a broad spectrum of accident scenarios;
- Accident management procedures;
- Component testing;
- Scaling issues;

— Database generation to support the licensing process;
— Code assessment and validation.

2.2.2.5. HLM pump and corrosion/erosion studies: Availability

The main pump is a critical component for LFR development, due to the physical properties of the coolant which flows inside the system and the relevance of the component in relation to reactor safety. Indeed, the main coolant pumps in nuclear technology are nuclear grade components.

The physical properties of the coolant and the critical relevance of the main pump to reactor safety refer to a process of rigorous manufacturing quality assurance for those components that are especially critical to reactor safety. Notwithstanding this, postulated initiating events in safety analysis and licensing refer to the malfunction of main coolant pumps (e.g. single or multiple main coolant pump failure, locked rotor and shaft seizure).

The importance of pump safety implies that experimental investigations are needed on materials, properties of the mechanical parts (i.e. impeller, bearings and housing), performance tests and reliability of the component. The R&D activities related to the pump impeller material have already been outlined. Other tests include:

— Bearing qualification tests;
— Integral pump tests addressing pump performances and long term reliability.

2.2.2.6. Instrumentation: Operation

LFRs impose higher requirements for instrumentation due to the higher thermal loads, higher temperatures, high fast neutron flux, corrosion effects and the opacity of liquid metals. These harsh environmental conditions in reactors, which can be even higher than the operational temperatures that need to be detected and recorded reliably in case of abnormal operation up to accident conditions, significantly limit the choice of instrumentation.

A common requirement for an LFR reactor design is therefore the understanding and development of materials and structures capable of functioning reliably for a long time in the harsh environment described above. Moreover, the position of instruments in a system influences their performance.

For most of the parameters to be measured, technical solutions for out-of-pile conditions are available or are currently under development and qualification. For in-pile conditions, the improvement and qualification of instrumentation has to be initiated and promoted, especially with respect to ageing and size.

2.2.2.7. Neutronics

Concerning neutronic issues related to LFR/ADS development, the following issues need to be addressed:

— Validation measurements for nuclear data improvement:
  • Threshold processes — (n, n′) and (n, xn) reactions;
  • MA cross-sections.
— Validation measurements for licensing and operation:
  • Uncertainty reduction on cross-sections (Pb-MA-Pu-241, Pu-242, in high energy range <1 MeV);
  • Determination of flux gradients in fast spectrum;
  • Reactivity effects (voiding of HLM coolant in mixed oxide core, secondary scram system).

The above areas and topics were identified for zero power reactor experiments (such as MASURCA and VENUS).

Concerning data libraries and neutronic code validation, the task includes experiments producing benchmarking data for:

— Validation and improving minor actinide cross-sections;
— Non-elastic neutron scattering thresholds;
— Libraries for innovative materials (SiC, ZrC, Zr_3Si_2 and others).

Concerning operational and control issues, the following need to be addressed carefully:

— Determination of flux peaks and gradients;
— Absorber and reflector worth;
— Degraded geometry studies.

3. OVERVIEW OF EXPERIMENTAL ACTIVITIES IN SUPPORT OF LMFNSs

3.1. SPECIFIC INTEREST FOR SODIUM COOLANT

Transporting thermal energy from the fast reactor core to the turbine requires a high level of reliability. The total heat transfer surface and mass of the IHXs and the steam generators in a fast reactor are less than those of the steam generators in a PWR. This is achieved through higher temperature differences across the heat exchanger and higher heat transfer coefficients due to the better heat transfer properties of liquid sodium.

Sodium has a relatively high boiling temperature (882°C), so the cooling system can operate at near atmospheric pressure and, therefore, the vessels and heat exchanger tubes can have thinner walls. Sodium is non-corrosive to structural materials used in the reactor and its density and viscosity are in the same range of values as water, allowing for the use of matured water coolant technology for the development of sodium cooled systems (e.g. primary pumps). Thus, sodium cooled systems possess unique parameters providing for superior reliability, operability, maintainability and long lifetime. These characteristics facilitate reducing the life cycle costs of SFRs. However, the strong chemical activity of sodium with air and water poses the risk of explosion and fire, which has to be managed by design provisions and operating procedures.

3.2. SPECIFIC INTEREST FOR HEAVY LIQUID METAL COOLANTS

The appeal of the LBE alloy as a coolant is due to its moderate melting point (125°C) and high boiling point (1638°C), practically eliminating the possibility of its boiling onset in areas with high temperatures, and very low chemical activity with respect to air, water and steam, thereby preventing any risk of explosion and fire. Low working pressure in the circuit increases the reliability and safety of the components, simplifies the design and manufacturing technology and significantly facilitates operation of the primary system components. In designing NPPs for nuclear submarines [16], the benefits of LBE outweigh its drawbacks, which include corrosion and erosion activity with respect to structural materials as well as polonium accumulation under radiation.

The interest of designers in larger NPPs with fast reactors using liquid lead is due to its low chemical reactivity with respect to water and air, its high boiling point (1745°C) and the insignificant rate of polonium accumulation under radiation. In addition to the drawbacks of LBE, the lead melting temperature is higher (327°C).

Both LBE and lead coolants dissolve oxygen. By carefully controlling the oxygen concentration in the liquid metal, stable oxide films can be formed on the steel surface [17]. The oxide films separate the steel from the corrosive liquid metal, and then the heavy direct dissolution of the steel is inhibited. Corrosion of stainless steels by liquid lead and LBE can be significantly mitigated by applying oxygen control technology. The corrosion inhibition efficiency of the technology depends not only on the oxygen concentration in the liquid but also on the type of structural steels and the system operating conditions. The low concentration of oxygen causes the oxide film on steel cladding to vanish and causes corrosion, while high oxygen concentration results in oxide buildup in the coolant. To maintain the protective films formed through reactions between oxygen and steel constituents and to avoid contaminating the liquid, the oxygen concentration has to be controlled instantaneously in a certain range. The operational limits are narrower for pure lead, thus making the oxygen control system in lead coolant more sophisticated. The oxygen control technology, including oxygen measurement and control methods, has been well developed. However, more studies are needed to determine the reasonable operating range of oxygen concentration in a non-isothermal loop system.

3.3. OVERVIEW AND OUTLOOK OF EXPERIMENTS IN SUPPORT OF LMFNSs

As noted, the acronym LMFNS is defined in this compendium as a common term for LMFNS SFRs and LFRs, both critical and subcritical, including LFR/ADS FNSs. An overview of experimental studies that have been performed or still need to be performed at existing facilities in support of LMFNS development is presented below.

3.3.1. Experimental studies at facilities in support of SFRs

The scientific basis for implementation of sodium coolant technology has been derived from more than fifty years' experimental work. Comprehensive studies on thermohydraulics, mechanisms of turbulent heat exchange, boiling and condensation, physical chemistry, technology and corrosion of structural materials in sodium have been carried out at experimental facilities with circulating sodium. Special attention was given to techniques and measurement technology, including the development of the unique detectors for measuring the temperature and velocity of the coolant.

Detailed data on temperature profiles and heat exchange have been obtained and used in reactor core calculations. During hydrodynamic studies of fuel assemblies and reactor flow sections, maximum attention has been paid to the measurements of velocity profiles, tangential stresses and flow turbulent characteristics. Detailed thermohydraulic studies of the full scale reactor core models — including possible overlapping of various core elements, emerging asymmetrical displacement and deformation of fuel pins and presence of counter flows — have been carried out [18, 19].

As a result of comprehensive studies of physical chemistry and sodium coolant technology [20], data have been obtained on various impurities in the sodium coolant, their equilibrium concentrations, solubility of oxygen, hydrogen and carbon, as well as on the kinetics of the reactions occurring in such systems. On the basis of these data and as a result of a study on the corrosion of structural materials, the admissible content of impurities in sodium and in protective gas has been determined. The concentrations of oxygen, carbon, hydrogen and nitrogen impurities are limited by corrosion processes and, for gaseous fission products (Cs, Sr), by the radiation environment.

These results confirm the low rate of corrosion of structural materials in sodium at equilibrium concentrations of oxygen and hydrogen during normal operation. The characteristics of the leak before break phenomenon (i.e. self-growing micro and small leaks due to a failure of the tube through which water flows into sodium as well as their transition into a large leak) have been studied. Requirements for the leakage detection systems have been established. Various methods of cleaning contaminants from the coolant have been studied [21, 22]. Cold and hot (getter) traps have been recommended for the practical applications, and carbon traps have been recommended for purification from caesium. Various designs of coolants' sample analysis techniques for hydrogen, oxygen and various forms of carbon as well as metal impurities have been developed and tested at the facilities.

Patterns of sodium burning, release of aerosol products and their transport in the building and into the environment have been also studied. The tools and localization systems for suppressing combustion and capturing aerosol products, protecting systems of concrete structures, methods and means for cleaning and decontaminating the equipment, and building after sodium burning have been developed.

The first objective of R&D studies on SFRs is the reactor itself. The main directions of experimental studies on sodium coolant in fast reactors that have been performed and those that still need to be carried out at existing facilities can be summarized in the following subsections [18–23].

3.3.1.1. Thermohydraulics

The main directions of experimental studies related to thermohydraulics in sodium coolants can be summarized as follows:

— Obtaining the main parameters of heat transfer (empirical relations, constants) and temperature distribution of fuel pins for all conditions and operating modes (geometry change, power excursion, peaking factors, etc.);
— Retrieving data for local turbulent characteristics for liquid metal single-phase and two-phase channel flow and pool boiling, taking into account the influence of large scale eddy flows on the thermal stratification;
— Validating computer codes based on results and using those results to obtain constants required for models which take into account the materials, operation parameters and designs proposed for future NPPs.

3.3.1.2. Physical chemistry and sodium coolant technology

The main directions of experimental studies related to physical chemistry and sodium coolant technology can be summarized as follows:

— Obtaining data on solubility, diffusion and dispersion characteristics of complex heterogeneous liquid metal systems and their behaviour, taking into account the spontaneous nucleation of the crystalline phase from a supersaturated solution;
— Studying the effect of coagulation processes in the circulation loop (coagulation rate is determined by the hydrodynamics of circulating coolant);
— Studying mechanisms and kinetics of the formation and disintegration of heavy oxides and carbides in a non-isothermal circuit;
— Discovering the minimum allowable concentration of oxygen and other impurities in sodium and sodium–potassium alloy to secure the performance of structural materials;
— Improving methods and systems of sodium purification — suspended solids, radioactive impurities;
— Justifying regimes that, when placing cold traps in the reactor tank, exclude the accumulation of hydrogen in the cold traps of the primary circuit;
— Developing methods of trapping and reliable localization of tritium emitted during various manufacturing operations.

3.3.1.3. SFR safety

The main objective of SFR safety studies is to ensure the safety of the primary circuit and the reactor core. The following is deemed necessary for future SFR safety studies:

— Justifying the design conditions, excluding the formation of eddy flows in the outlet channel of the reactor core and on the surface of sodium pool (gas trapping), and passive circulation zones (stratification phenomenon, temperature fluctuations).
— Analysing the consequences of possible non-standard mode of operation (e.g. locks, triggering emergency core cooling systems, boiling) and developing accident mitigation measures.
— Rationalizing and developing new technical proposals to avoid the large external sodium systems in emergency heat removal systems and their other deficiencies.
— Optimizing the characteristics of air heat exchanger of the emergency heat removal systems of SFRs: it is necessary to justify the distribution of airflow at the inlet of the fuel assembly, check the sodium circulation resistance in parallel circuit systems to study the processes of heat transfer and thermal strength of the wire-wrapped rod bundles in non-stationary DECs.
— Examining new design solutions capable of ensuring reactor core cooling at the sodium boiling point, namely, the sodium plenum above the reactor core. It is required to determine the boundaries of unstable operation and to study the dynamics of the propagation of the boiling region in a fuel assembly.
— Testing passive safety systems proposed for protection against overheating of passive safety devices, triggered by reducing the flow rate through a passive safety system and by raising the temperature at its outlet.

The second important element is the steam generator. The substantiation of thermohydraulic regimes for steam generators and their system of automatic protection is a very important task, which involves the following:

- Developing and testing the materials and the overall design to improve the safety of a large steam generator, securing the sustained process of self-growing leakages and operative repair in case of water ingress in sodium;
- Validating experimentally, for advanced large block steam generators, the thermohydraulic regimes and testing the system of automatic protection;
- Continuing research to improve leak detection systems by applying various detection methods such as measuring concentrations, vibroacoustic and others.

3.3.1.4. Eliminating sodium fires and increasing hydrogen safety

The main directions of experimental studies related to eliminating sodium fires and increasing hydrogen safety in sodium coolants can be summarized as follows:

- Developing and experimentally validating measures and technical solutions that minimize leaks of sodium and its contact with the air;
- Paying particular attention to the 'leak before break' concept;
- Developing new aerosol filters for emergency ventilation systems triggered by sodium leakage and for the general ventilation system to improve the efficiency of aerosol containment systems.

3.3.1.5. Other

There is also a need to perform experimental validation of the solutions to:

- Extending the life of SFRs with a higher coolant temperature;
- Improving monitoring devices and technologies for cleaning sodium circuits;
- Decommissioning and reprocessing sodium.

Implementation of these tasks requires an appropriate experimental base and staff. Creating consolidated databases as a follow-up to these studies is deemed necessary.

3.3.2. Experimental studies at facilities in support of LFRs

Developments carried out in the last decade regarding ADSs as well as fast reactors evidenced lead coolant as an emergent technology potentially complying with all GEN-IV goals (with specific reference to nuclear cycle sustainability). Thanks to the high boiling point of lead, together with its inert chemical nature, its favourable neutronics and safety characteristics, LFR technology has attracted the attention of research organizations and industry as a credible alternative to the other fast neutron reactor technologies under R&D.

The most challenging issue to be thoroughly analysed is the compatibility of the structural materials and coolant chemistry handling. Important actions have been taken so far in the development of LFR technology [24].

Since fuel cladding material cannot be developed completely in a short time, most of the current designs foresee retaining the relevant design choices adopted for ELSY and ELFR [24], namely, to cautiously limit the bulk core outlet temperature to 480°C, adopt the already qualified material for cladding and improve it against HLM corrosion by using corrosion resistant coats. The absence of chemical reactivity in lead allows for placing the steam generator directly into the primary pool without any intermediate system. The adoption of this advantageous configuration requires, however, an accurate analysis of the consequences of an SGTR accident in terms of damage to the surrounding structures. R&D studies (both calculations and experiments) have been performed on this issue to implement provisions preventing and limiting the possible effects of a shock wave and preventing water vapour from entering the core.

Comprehensive thermohydraulic simulations of large components and plant configuration have been performed at large HLM experimental facilities. These simulations demonstrate the thermomechanical effects of thermal striping and stratification in order to optimize the design and materials.

There is no operating experience or feedback on reactors cooled by pure lead. About eighty reactor years of experience and feedback were accumulated during operation of LBE cooled reactors used for Alfa/Lira class submarines and land based facilities in the former Soviet Union. The related feedback as well as experience from licensing of these reactors is, however, not easily available.

On the basis of analyses performed by the GIF Risk and Safety Working Group [24], the main safety features and issues for LFRs are discussed in the following subsections.

3.3.2.1. Materials and coolants

The main safety features and issues for LFRs related to materials and coolants can be summarized as follows:

— Molten lead is corrosive and becomes oxidized if oxygen concentration is not controlled.
— Molten lead might erode structural materials, allowing metallic impurities produced by corrosion to be transported in the primary system.
— Lead vapours are toxic.
— A large specific weight of lead and total primary inventory might, in the case of external excitations, challenge the structural integrity of systems or components.
— Large quantities of coolant in the main vessel of pool LMFNSs (both SFRs and LFRs) may lead to complex flow patterns and interactions between the coolant and structures.
— Lead is opaque.

The specific aspects related to the main safety functions, such as reactivity control, heat removal and confinement of radioactivity, are outlined in the following subsections.

3.3.2.2. Reactivity control

Specific aspects of the main safety functions related to reactivity control can be outlined as follows:

— Ruptures of steam generator tubes might lead to over-pressurization of the primary circuit, sloshing and steam/water entrainment, resulting in a positive reactivity insertion.
— The lead density reactivity coefficient might be positive in some core regions.
— Loss of core geometry (core compaction) might lead to a positive reactivity insertion and power increase.

3.3.2.3. Heat removal safety aspects

Specific aspects related to heat removal safety can be outlined as follows:

— Lead has a high melting point (327°C) with a potential for coolant solidification. Mechanical stresses might be exerted on structures during melting if a proper melting sequence is not applied.
— Accumulation of corrosion products might lead to coolant blockages.

3.3.2.4. Confinement of radioactivity

Specific aspects related to confinement of radioactivity can be outlined as follows:

— Corrosive properties of molten lead could challenge confinement barriers, in particular, the cladding of the fuel pins.
— Proper design can turn what was previously considered a serious drawback (the high melting point of lead) into an advantage, whereby frozen lead can effectively seal potential cracks and other faults, thus enhancing the confinement barriers. This was shown in the recent ELSY project in the EU [24].

3.3.3. Promote the experience — outlook for LFR development

Promoting LFR development involves the following:

— Identifying innovative nuclear R&D activities that require research reactor support;
— Identifying existing (or soon to be operational) research reactor facilities capable of supporting innovative nuclear development;
— Quantifying the capabilities of the identified facilities within the context of the required research support.

It should be noted that:

— A smaller vessel (less than 10 m for a 600 MW(e) reactor), robust to seismic loads, is possible.
— About a 1.5 m level difference between the cold and hot plenum at normal power operation is sufficient to feed the core and simplifies the primary system configuration.
— Refuelling machines operating on gas would resolve the issue related to in-vessel handling in an opaque medium at high temperature.
— No component is connected to the reactor vessel, making it possible to replace all components, including the cylindrical inner vessel, through reactor roof penetrations, eliminating the need for repair under hot lead.
— Thanks to the hard neutron spectrum, most threshold fission reactions of fissionable isotopes are effectively triggered, determining low minor actinide equilibrium concentrations in the fuel.

4. EXPERIMENTAL FACILITIES IN SUPPORT OF LMFNSs

4.1. INTRODUCTION

The important goal of this compendium is to consider LMFNSs and to highlight the required experimental activities in support of them. These activities are driven by the development of innovative techniques, concepts and components in different fields: ECS, instrumentation for continuous monitoring, ISI&R, core design, fuel handling, thermohydraulics, severe accidents, large flow electromagnetic pumps, materials, and the like. In many cases, the development process relies on several steps: fundamental studies; simulation; and V&V&Q of innovative options. For V&V&Q, experimental tests are required. Thus, the development of LMFNSs should be supported by experimental activities using experimental facilities.

Since the requirements are very specific to SFRs and LFRs, the facilities in support of LMFNSs could require large investments. It is very important to clearly substantiate the needs of the different experimental programmes. It is of great interest to identify whether any similar facility already exists in other countries. In this case, some other considerations should be addressed, such as a facility's usability by others than the owner, its availability, and the possibility of adapting it to a specific request of an experiment.

As was noted in Section 1.4, the main component of the compendium consists of individual papers (profiles) providing a technical description of the facilities in support of the development and deployment of innovative LMFNSs, including their specific features for utilization. The profiles have been provided by research organizations and other relevant institutions in IAEA Member States following a template that provides structured information on facilities that are currently being designed, are under construction or are already operating in support of the development and deployment of liquid metal cooled (sodium, lead and lead–bismuth) FNSs, both critical and subcritical.

Facilities that will be operational after 2020, even if they are currently under design or construction, are considered beyond the scope of this compendium. The cut-off for identifying the facilities to be considered in this compendium was June 2015.

The profiles are collected in alphabetical order by country in two folders on CD-ROM: the first folder contains the profiles of facilities in support of SFRs, and the second contains those in support of LFRs. An overview of facilities by country is presented in Table 2.

Nineteen institutions from 14 countries and the European Union, which contributed to the LMFNS compendium project, provided data on 79 facilities in support of SFR development and 72 facilities in support of LFR development. From the total number of 150 facilities, 14 are designed for dual application, meaning they can be used for the development of both sodium and lead/LBE FNSs.

TABLE 2. FACILITIES IN IAEA MEMBER STATES IN SUPPORT OF THE DEVELOPMENT AND DEPLOYMENT OF INNOVATIVE LMFNSs

Country	Purpose/ Number of facilities		Cross-cut (dual use possible)	Total by country Na based plus Pb based facilities
	Na based	Pb based	Na and Pb/LBE	
Belgium		15		15
China	8	7		15
Czech Republic		4		4
EU		1		1
France	19	2	2	21
Germany	5	11		16
India	14			14
Italy		10		10
Japan	6	6		12
Korea, Republic of	4	1		5
Latvia	3	1		4
Russian Federation	11	10	(8)	21
Spain		2		2
Sweden		1		1
USA	9	1	(4)	10
Total by purpose	79	72	(14)	151

LBE — lead–bismuth eutectic; Na — sodium; Pb — lead.

The R&D requirements discussed in the previous section led to the following considerations about experimental needs:

— The experimental needs are varied and require a large number of different operating conditions and operating fluids.
— As a large number of experimental facilities exist all over the world, this response appears to be feasible. In some cases, adaptations to their specificities could be required.
— As the investment cost for such facilities is significant, the opportunity provided by this compendium to share the use of some of them could be very helpful.
— To respond to some specific needs, new facilities have to be designed and commissioned with consideration of:
  • Flexibility to adjust their performance at an early design stage to permanently evolving R&D needs;
  • Desired design features allowing simplification of their operation.

The facilities covered in this compendium are categorized into two main groups:

(1) Experimental facilities devoted to SFR system development;
(2) Experimental facilities devoted to LFR system development.

In addition to the facilities in support of SFRs and LFRs, there are a few facilities in support of both designs. These facilities are capable of supporting studies on DBAs and thermohydraulic experiments. In this compendium, these cross-cutting facilities are reported in both groups (i.e. SPRUT, BFS-1, BFS-2).

Neutron sources (i.e. nELBE, TEF-T) available for supporting LMFNS development have been considered and categorized in this compendium, as well as zero power reactors (i.e. MASURCA, VENUS-F). Facilities related to the development of proton accelerators for ADSs were not considered.

Research reactors are considered beyond the scope of this compendium (see Refs. [25, 26]), as are general facilities and laboratories (i.e. creep laboratories, metallographic laboratories, etc.), post-irradiation examination facilities, fuel and fuel cycle facilities.

Tables 3 and 4 in Sections 4.3 and 4.4 provide an overview of possible applications of the experimental facilities in support of SFRs and LFRs, respectively. These facilities are much diversified in their use and some of them cover several application areas (research fields). Their features need to be analysed in detail in order to have a clear view of their capabilities. Indeed, some categories, such as instrumentation or thermohydraulics, are very broad, and that can correspond to different requirements and test sections (e.g. using water or sodium as a coolant in representative conditions).

4.2. FACILITY PROFILE TEMPLATE

For each facility being considered, a suitable template is proposed, aiming to improve readability while providing enough information and detail to understand the capabilities of the facility and the background of the research group that is managing and operating the research infrastructure.

For each facility, the following is outlined:

— General information: coolant of the facility, its location, operator and the contact points of the facility.
— Status of the facility:
  - In operation;
  - Standby;
  - Under construction;
  - Under design;
  - Planned, and operational by 2020.

As already mentioned, the facilities are first categorized by the coolant of the LMFNSs. A second level categorization is made considering the most relevant research field of the facility:

— Main research field(s);
— Zero power facility for V&V and licensing purposes;
— DBA and DEC;
— Thermohydraulics;
— Coolant chemistry;
— Materials;
— Systems and components;
— Instrumentation and ISI&R.

Many facilities include more research fields than are outlined in the template, being designed and operated as multipurpose facilities. For each facility a main research field is identified, allowing a categorization at the second level:

— *Technical description:* A short description of the facility is provided, outlining capabilities and available test sections. Schemes and piping and instrumentation diagrams are reported in 3-D, accompanied by relevant pictures, if available. The possibility of hosting radioactive material is also outlined in this field.
— *Completed experimental campaigns:* The main experimental results achieved at the facility are described, clearly identifying the capabilities of the infrastructure and the fields of application.
— *Planned experiments (including time schedule):* The most recent refurbishment of the facility, identifying the immediate frame of collaboration, is outlined.
— *Training activities:* Education and training capabilities and practical information for the implementation of these activities are described.
— *References:* A short list of papers and publications useful for users to get more detailed information about the infrastructure is presented.

4.3. EXPERIMENTAL FACILITIES IN SUPPORT OF SFRs

In this compendium, 79 facilities in support of SFR development operating primarily with water, air or sodium are presented. An overview of these SFR facilities by their most relevant research field (main application) is presented in Table 3.

From the total number of 79 facilities located in 9 countries:

— 57 facilities are operational and available for research;
— 7 facilities are currently in standby mode;
— 10 facilities are under construction or upgrade/modification;
— 5 facilities are under design;
— 10 facilities can also be used for LFR development.

For detailed information on these facilities, see the SFR profiles on the CD-ROM attached to this report.

4.4. EXPERIMENTAL FACILITIES IN SUPPORT OF LFRs

In this compendium, 72 facilities in support of LFR development are reported on and classified according to their most relevant research fields (main application). In Table 4, a general overview of the LFR facilities is presented.

Of the total number of 72 facilities located in 13 countries and in the EU:

— 55 facilities are operational and available for research;
— 7 facilities are currently in standby mode;
— 6 facilities are under construction or upgrade/modification;
— 3 facilities are under design;
— 12 facilities can be also used for SFR development.

It should be noted that, for any research field (apart from the zero power facilities for VV&Q and licensing purposes), several facilities are currently available, attesting to the great interest worldwide in LFR development, which involves mainly China, the Russian Federation and the European Union.

For detailed information on these facilities, see the LFR profiles on the CD-ROM (or in the retrieval system) attached to this compendium.

TABLE 3. FACILITIES IN SUPPORT OF SODIUM COOLED REACTOR BASED FAST NEUTRON SYSTEMS (SFRs)

Profile		Country	Facility	Main application							
				Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
SFR	1	China	CEDI		■	■					
SFR	2	China	ESPRESSO		■	■					
SFR	3	China	FRIYG-l			■					
SFR	4	China	HTMTSL					■			
SFR	5	China	HTTCSL					■			
SFR	6	China	MSSPD				■				
SFR	7	China	SIPHON		■	■					
SFR	8	China	TSBS			■		■			
SFR	9	France	BACCARA								■
SFR	10	France	CARNAC				■				
SFR	11	France	CHEOPS–NADYNE Esa			■		■	■		
SFR	12	France	CHEOPS–NAIMMO Esa			■			■	■	
SFR	13	France	CHEOPS–NSET Esa			■			■		
SFR	14	France	CORONA					■			
SFR	15	France	DIADEMO Na			■			■		
SFR	16	France	DOLMEN						■	■	
SFR	17	France	FUTUNA 2				■	■		■	
SFR	18	France	IRINA			■			■	■	
SFR	19	France	LIQUIDUS					■		■	
SFR	20	France	MASURCA								■
SFR	21	France	MECANA							■	

TABLE 3. FACILITIES IN SUPPORT OF SODIUM COOLED REACTOR BASED FAST NEUTRON SYSTEMS (SFRs) (cont.)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
SFR 22	France	PEMDYN						■	■	
SFR 23	France	PENELOPE						■	■	
SFR 24	France	PLATEAU			■			■	■	
SFR 25	France	PLINIUS 2		■			■			
SFR 26	France	SUPERFENNEC				■			■	
SFR 27	France	VKS2							■	
SFR 28	Germany	ALINA			■		■	■	■	
SFR 29	Germany	DRESDYN			■				■	
SFR 30	Germany	KASOLA		■	■		■	■	■	
SFR 31	Germany	NATAN			■		■	■	■	
SFR 32	Germany	SOLTEC		■	■		■	■	■	
SFR 33	India	500 kW sodium loop			■			■	■	
SFR 34	India	BIM					■		■	
SFR 35	India	INSOT creep loop					■			
SFR 36	India	INSOT fatigue loop					■			
SFR 37	India	LCTMF						■	■	
SFR 38	India	LCTR			■			■	■	
SFR 39	India	LEENA							■	
SFR 40	India	SADHANA			■			■		
SFR 41	India	SAMRAT								
SFR 42	India	SGTF			■				■	

TABLE 3. FACILITIES IN SUPPORT OF SODIUM COOLED REACTOR BASED FAST NEUTRON SYSTEMS (SFRs) (cont.)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
SFR 43	India	SILVERINA sodium loop					X			
SFR 44	India	SOWART		X			X		X	
SFR 45	India	Subassembly hydraulic test rig			X			X		
SFR 46	India	TSTF								
SFR 47	Japan	AtheNa		X	X			X	X	
SFR 48	Japan	CCTL			X					
SFR 49	Japan	MELT		X	X					
SFR 50	Japan	PLANDTL			X			X		
SFR 51	Japan	SAPFIRE		X		X				
SFR 52	Japan	SWAT		X		X	X	X		
SFR 53	Korea, Republic of	ITSL						X	X	
SFR 54	Korea, Republic of	SOFUS							X	
SFR 55	Korea, Republic of	STELLA-1			X			X	X	
SFR 56	Korea, Republic of	STELLA-2		X	X			X		
SFR 57	Latvia	RIGADYN		X	X			X		
SFR 58	Latvia	TESLA			X		X	X	X	
SFR 59	Latvia	ST-300			X		X	X	X	
SFR 60	Russian Federation	6B		X	X				X	X
SFR 61	Russian Federation	AR-1		X	X					
SFR 62	Russian Federation	B-2			X	X				X
SFR 63	Russian Federation	BFS1		X	X					X
SFR 64	Russian Federation	BFS2		X	X					X

TABLE 3. FACILITIES IN SUPPORT OF SODIUM COOLED REACTOR BASED FAST NEUTRON SYSTEMS (SFRs) (cont.)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
SFR 65	Russian Federation	Pluton		■						
SFR 66	Russian Federation	Protva-1			■		■	■	■	
SFR 67	Russian Federation	SGDI			■					■
SFR 68	Russian Federation	SGI		■						■
SFR 69	Russian Federation	SPRUT			■					■
SFR 70	Russian Federation	V-200		■	■					■
SFR 71	USA	ALEX			■	■	■	■	■	
SFR 72	USA	Creep lab–MPM					■			
SFR 73	USA	Fracture mechanics laboratory					■			
SFR 74	USA	Hydraulic lab–MPM					■			
SFR 75	USA	MCF					■			
SFR 76	USA	Material processing facility					■			
SFR 77	USA	METL			■	■	■	■	■	
SFR 78	USA	SNAKE			■	■				
SFR 79	USA	UW sodium loops 1 and 2				■	■		■	
Total by main application			0	14	42	11	29	30	34	10

DBA — design basis accident; DEC — design extension condition; ISI&R — in-service inspection and repair.

TABLE 4. COMPENDIUM PROFILES LFR: FACILITIES IN SUPPORT OF HEAVY LIQUID METAL COOLED FNSs (LFRs)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
LFR 1	Belgium	COMPLOT			■			■	■	
LFR 2	Belgium	CRAFT V2				■	■		■	
LFR 3	Belgium	E-SCAPE		■	■				■	
LFR 4	Belgium	Helois III				■				
LFR 5	Belgium	HLM		■	■				■	
LFR 6	Belgium	Liliputter				■		■	■	
LFR 7	Belgium	LIMETS1					■			
LFR 8	Belgium	LIMETS2					■			
LFR 9	Belgium	LIMETS3					■			
LFR 10	Belgium	LIMETS4					■			
LFR 11	Belgium	MEXICO				■				
LFR 12	Belgium	MYRRHABELLE			■					■
LFR 13	Belgium	SHAKESPEARE		■						■
LFR 14	Belgium	RHAPTER						■	■	
LFR 15	Belgium	VENUS-F	■						■	
LFR 16	China	CLEAR-0	■					■	■	
LFR 17	China	CLEAR-S		■	■	■	■	■	■	
LFR 18	China	KYLIN II M				■	■			
LFR 19	China	KYLIN II S		■				■	■	
LFR 20	China	KYLIN II TH FC			■	■		■	■	
LFR 21	China	KYLIN II TH MC			■	■		■	■	
LFR 22	China	KYLIN II TH NC			■				■	

TABLE 4. COMPENDIUM PROFILES LFR: FACILITIES IN SUPPORT OF HEAVY LIQUID METAL COOLED FNSs (LFRs) (cont.)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
LFR 23	Czech Republic	CALLISTO					■		■	
LFR 24	Czech Republic	COLONRI I				■	■			
LFR 25	Czech Republic	COLONRI II				■	■			
LFR 26	Czech Republic	MATLOO Pb				■	■			
LFR 27	EU	HLM EF					■		■	
LFR 28	France	BACCARA			■			■		■
LFR 29	France	MASURCA	■							■
LFR 30	Germany	CORELLA					■			
LFR 31	Germany	CORRIDA				■	■		■	
LFR 32	Germany	COSTA				■	■			
LFR 33	Germany	CRISLA					■			
LFR 34	Germany	ELEFANT			■		■		■	
LFR 35	Germany	FRETHME					■			
LFR 36	Germany	MINIPOT			■	■	■	■	■	
LFR 37	Germany	nELBE					■			
LFR 38	Germany	TELEMAT				■	■	■	■	
LFR 39	Germany	THEADES			■			■	■	
LFR 40	Germany	THESYS			■		■	■	■	
LFR 41	Italy	CIRCE–HERO		■	■	■	■	■	■	
LFR 42	Italy	CIRCE–SGTR		■	■			■	■	
LFR 43	Italy	GIORDI					■			
LFR 44	Italy	HELENA			■	■	■	■	■	
LFR 45	Italy	LECOR				■	■		■	

TABLE 4. COMPENDIUM PROFILES LFR: FACILITIES IN SUPPORT OF HEAVY LIQUID METAL COOLED FNSs (LFRs) (cont.)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
LFR 46	Italy	LIFUS5		■	■	■		■	■	
LFR 47	Italy	NACIE-UP		■	■			■	■	
LFR 48	Italy	RACHELE				■	■	■	■	
LFR 49	Italy	SIRIO		■	■			■	■	
LFR 50	Italy	SOLIDX		■	■					
LFR 51	Japan	FundamentalTest						■	■	
LFR 52	Japan	High temperature material				■	■			
LFR 53	Japan	JLBL-3			■			■		
LFR 54	Japan	JLBL-4			■				■	
LFR 55	Japan	TEF mock-up			■		■	■	■	
LFR 56	Japan	TEF	■		■		■	■		
LFR 57	Korea, Republic of	HELIOS			■	■	■		■	
LFR 58	Latvia	IPUL			■		■	■	■	
LFR 59	Russian Federation	B-2			■					■
LFR 60	Russian Federation	BFS-1	■	■						■
LFR 61	Russian Federation	BFS-2	■	■						■
LFR 62	Russian Federation	SGDI			■					■
LFR 63	Russian Federation	SGI		■	■					■
LFR 64	Russian Federation	SPRUT			■					■

TABLE 4. COMPENDIUM PROFILES LFR: FACILITIES IN SUPPORT OF HEAVY LIQUID METAL COOLED FNSs (LFRs) (cont.)

Profile	Country	Facility	Main application							
			Zero power	DBA and DEC	Thermohydraulics	Coolant chemistry	Materials	System and components	Instrumentation and ISI&R	Cross-cutting facility
LFR 65	Russian Federation	TT-1M								
LFR 66	Russian Federation	TT-2M								
LFR 67	Russian Federation	V-200								
LFR 68	Russian Federation	6B								
LFR 69	Spain	LINCE								
LFR 70	Spain	SFLM								
LFR 71	Sweden	TALL3D								
LFR 72	USA	DELTA								
Total by main application			6	15	34	25	37	28	34	12

DBA — design basis accident; DEC — design extension condition; ISI&R — in-service inspection and repair.

5. FUTURE DEVELOPMENT OF LMFNS FACILITIES

5.1. FUTURE DEVELOPMENT OF FACILITIES IN SUPPPORT OF SFRs

Sodium coolant is used in the primary and secondary circuits of most fast reactors. The third circuit is usually designed as a steam turbine cycle.

Intensive heat removal in the heat exchange apparatus of an NPP is achieved due to the thermal properties of sodium. Its disadvantage is its violent reaction with water, accompanied by the formation of alkali, hydrogen and heat. Thus, sodium water steam generators are the most critical elements that largely determine the performance of NPPs.

Extensive experience with operating sodium water steam generators has confirmed their efficiency. However, establishing a reliable steam generator design remains a priority.

Water leaks in sodium have occurred in virtually all steam generators of all NPPs with fast neutron reactors: Prototype Fast Reactor (PFR) (United Kingdom), Phénix (France), BN-350 and BN-600 (former USSR) and Enrico-Fermi (USA). The interaction of sodium with water was detected even in the steam generator with two-wall tubes at experimental reactor KNK-II (Germany). At the Phénix reactor, after nine years of operation, all the steam evaporation units had been replaced. At PFR, from the very beginning of its operation in 1974, numerous leaks of water to sodium were observed with the tube plate of U-shaped steam generators in welded pipe joints. Mainly for this reason, the average load factor of PFR was only 11%.

There are currently no structural materials that can guarantee sufficient stability in a zone where sodium reacts with water. Therefore, operating steam generators, even with only small leaks, is unacceptable. Sodium water steam generators should be equipped with leak detection systems, dehydration and drainage of sodium and inert gas filling steam generator cavities. This suite of systems is called the 'emergency protection system of the steam generator'. All systems should follow the established algorithm automatically.

Construction of a large scale thermohydraulic facility [27, 28] is deemed necessary to assess the emergency protection system for SFR steam generators in case of water (steam) leakage into the shell side (sodium circuit). Seal failure of the heat transfer surface in steam generator modules can be simulated by injecting water (steam) into the inter-pipe space of modules and can be fixed by the leak monitoring system. In addition, studies to define the effective parameters of detection systems and how best to minimize pollution and design damage at water (steam) leaks, as well as hydrodynamic and temperature effects, can be performed at such a facility.

5.2. FUTURE DEVELOPMENT OF FACILITIES IN SUPPORT OF LFRs

According to the previous overview, beginning with the existing facilities already mentioned in this compendium, to demonstrate LFR (up to Technological Readiness Level 7 — TRL7), the construction of the following facilities would be necessary:

- An experimental facility for corrosion testing of materials in lead/LBE environments at high temperature (650°C) and high velocity (2 m/s).
- Experimental facilities for studies of LME, fatigue, creep, stress corrosion cracking up to 650°C and in HLM (with oxygen control systems).
- An experimental facility for SGTR having an interaction volume meaningful for testing a significant portion of steam generators and representative of 'direct' extrapolation to reactor vessel dimensions. This facility will provide a realistic reproduction of the tube rupture (i.e. refurbishment of CLEAR-S).
- An experimental facility for heat exchange components: steam generators and the DHR system. The facility will permit testing safely, with components having a total heat exchange power of 2 to 10 MW for LFRs.
- A new facility, or renovation of an existing one, to investigate the pool thermohydraulics of HLM. The alternative option may be selected following an upgrade of the instrumentation of the existing facilities (i.e. E-SCAPE, CIRCE, CLEAR-S).
- Facilities (e.g. COMPLOT) to perform full scale experimental qualification of the fuel elements in a hydraulic flow through the core in normal and accidental conditions. Such facilities should include multipurpose functions to support the designer with full scale basic tests (e.g. operability of the handling machine, operability and insertion speed of the control rods, capability of cooling fuel assemblies during refuelling) to test relevant mechanical components before implementing them in the demonstration reactors.
- A facility for studying the origins of core damage events and investigating severe accidents.
- More experimental facilities to perform integral qualification tests of the main coolant pump (i.e. CLEAR-S). Equipment should include qualification instruments for thermohydraulics, vibration dynamics, mechanical forces and overall performance.
- An experimental facility for corrosion tests, having a larger diameter test section than those currently available. This facility should be able to test the performances of a specific number of pins simultaneously with a coolant velocity of up to 2 m/s.
- A coolant chemistry facility for LFR design to investigate the coolant and a facility for studying fission gas release in the cover gas.
- Test facilities for investigating fuel coolant interaction (basic chemistry) and fuel dispersion in the primary system. The implementation of a suitable hot cell is needed.
- Test facilities for seismic testing.

Demonstration reactors BREST-300, ALFRED, MYRRHA and CLEAR-I are currently under development and will be created with an aim to:

— Demonstrate (TRL7) the HLM nuclear system technology;
— Achieve high safety standards and enhance non-proliferation resistance;
— Assess the economic competitiveness of LFR technology, including high load factors;
— Demonstrate better use of resources by closing the fuel cycle;
— Validate selection of materials.

The above mentioned demonstration reactors are also designed to confirm that the newly developed and adopted materials — both structural material and innovative fuel material — are able to sustain high fast neutron fluxes and high temperatures.

6. CONCLUSIONS

In this compendium, comprehensive information on existing and future experimental facilities able to support innovative LMFNS systems is presented.

Twenty institutions from 14 countries and the EU contributed to the LMFNS project by providing their input. In total, this compendium covers information about 151 LMFNS experimental facilities. The thermal power of these facilities ranges from 0 to 1 MW.

Despite the different maturity levels of sodium and lead/LBE technologies, and acknowledging greater practical experience with SFRs, there is continued commitment in the nuclear research community to developing both SFR and LFR FNSs. There are approximately the same number of facilities in support of SFR and LFR development and deployment: 79 for SFRs and 72 for LFRs.

There are 57 facilities that are operational and available for research for SFRs and 55 for LFRs. There are currently seven facilities in support of SFRs and seven supporting LFRs on standby mode. Ten facilities in support of SFRs and six for LFRs are under construction or are being upgraded of modified. There are five facilities for SFRs and three facilities for LFRs currently under design.

Fourteen of the 151 facilities are designed for dual purpose application, meaning they can be used for experiments on both sodium and lead/LBE FNSs.

Identifying the existing experimental infrastructures, as well as the new experimental facilities, based on the recognized R&D needs in Member States with fast reactor programmes is considered a priority. The IAEA actively promotes the harmonization of these efforts at the international level.

Taking into account the different development levels of sodium and lead/LBE technologies, the following future R&D needs have been identified by the participants of the LMFNS study:

— For SFR related activities, the primary task is comprehensively studying sodium–water interaction in steam generators, which are the most critical elements of SFR design. Construction of a large scale thermohydraulic facility is deemed necessary to assess the emergency protection system for SFR steam generators in cases of water (steam) leakage into the shell side (sodium circuit).
— For LFR related activities, several tasks have been outlined, including construction of new experimental facilities to test corrosion of materials in a lead/LBE alloy environment as well as to study LME, fatigue, creep, stress corrosion cracking up to 650°C, SGTR, heat exchange, etc.

Since the requirements for future LMFNSs are very specific, and facilities in support of SFRs and LFRs could represent a large investment, it is important to clearly substantiate the need for different experimental programmes. And it might be helpful to identify, if any similar facilities already exist in other countries, whether they are accessible and available for research by other than the facility owner and whether it would be possible to adapt the facility to the specific demands of an experiment.

In this compendium, detailed information on the facilities is presented in the form of individual papers (profiles) contained on an attached CD-ROM. This initiative will also produce a living database on these experimental facilities, to be regularly updated (first update to be published by 2019) by the IAEA and interested Member States.

This compendium will serve as a tool to facilitate cooperation between organizations with active programmes on LMFNSs. It is expected that it will enhance the utilization of existing experimental facilities, identify further R&D needs towards further deployment of fourth generation fast reactors and assist in the education and training of engineers in the field of liquid metal coolant technology.

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ABBREVIATIONS

ADS	accelerator driven system
CEFR	China experimental fast reactor
DBA	design basis accident
DEC	design extension condition
DHR	decay heat removal
ECS	energy conversion system
ELFR	European lead fast reactor
ELSY	European lead cooled system
FBTR	fast breeder test reactor
FNS	fast neutron system
FRKP	fast reactor knowledge preservation
GEN-III	Generation III
GEN-IV	Generation IV
GIF	Generation IV International Forum
HLM	heavy liquid metal
IHX	intermediate heat exchanger
INIS	International Nuclear Information System
ISI&R	in-service inspection and repair
LBE	lead–bismuth eutectic
LFR	lead cooled fast reactor
LME	liquid metal embrittlement
LMFNS	liquid metal cooled fast neutron system
NPP	nuclear power plant
PFBR	prototype fast breeder reactor
PFR	prototype fast reactor
PWR	pressurized water reactor
R&D	research and development
SFR	sodium cooled fast reactor
SGTR	steam generator tube rupture
TRL	technological readiness level
VV&Q	verification, validation and qualification

Annex I

PARTICIPANTS OF THE LMFNS PROJECT

The following 20 institutions from 14 countries and the European Union have contributed to the LMFNS project by providing their input to this compendium.

Belgium	SCK•CEN	Belgian Nuclear Research Centre
China	CIAE	China Institute of Atomic Energy
China	INEST-CAS	Institute of Nuclear Energy Safety Technology of Chinese Academy of Sciences
Czech Republic	CVŘ	Research Centre Řež
EU	JRC	Joint Research Centre — the European Commission’s science service
France	CEA	French Alternative Energies and Atomic Energy Commission
Germany	KIT	Karlsruhe Institute of Technology
Germany	HZDR	Helmholtz-Zentrum Dresden-Rossendorf
India	IGCAR	Indira Gandhi Centre for Atomic Research
Italy	ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
Japan	JAEA	Japan Atomic Energy Agency
Korea, Republic of	KAERI	Korea Atomic Energy Research Institute
Korea, Republic of	KINS	Korean Institute of Nuclear Safety
Latvia	IPUL	Institute of Physics of the University of Latvia
Russian Federation	IPPE	Institute of Physics and Power Engineering — State science centre of the Russian Federation
Spain	CIEMAT	Spanish Research Centre for Energy, Environment and Technology
Sweden	KTH	KTH Royal Institute of Technology
USA	ANL	Argonne National Laboratory
USA	LANL	Los Alamos National Laboratory
USA	ORNL	Oak Ridge National Laboratory

Annex II

CONTENTS OF THE ATTACHED CD-ROM

The case studies in the Annex have been prepared from the original material as submitted for publication and have not been edited by the editorial staff of the IAEA. The case studies remain true to the original reports submitted by the Member State.

The contents of the CD-ROM are available on the IAEA Publications web site. To download, please visit www-pub.iaea.org/books/IAEABooks

Profiles of sodium cooled fast reactors (SFRs) and lead cooled fast reactors (LFRs) provide a thorough technical description of the facilities, including their specific features for utilization.

SFR 1: CEDI, China

SFR 2: ESPRESSO, China

SFR 3: FRIYG-1, China

SFR 4: HTMTSL, China

SFR 5: HTTCSL, China

SFR 6: MSSPD, China

SFR 7: SIPHON, China

SFR 8: TSBS, China

SFR 9: BACCARA, France

SFR 10: CARNAC, France

SFR 11: CHEOPS–NADYNE Esa, France

SFR 12: CHEOPS–NAIMMO Esa, France

SFR 13: CHEOPS–NSET Esa, France

SFR 14: CORONA, France

SFR 15: DIADEMO Na, France

SFR 16: DOLMEN, France

SFR 17: FUTUNA 2, France

SFR 18: IRINA, France

SFR 19: LIQUIDUS, France

SFR 20: MASURCA, France

SFR 21: MECANA, France

SFR 22: PEMDYN, France

SFR 23: PENELOPE, France

SFR 24: PLATEAU, France

SFR 25: PLINIUS 2, France

SFR 26: SUPERFENNEC, France

SFR 27: VKS2, France

SFR 28: ALINA, Germany

SFR 29: DRESDYN, Germany

SFR 30: KASOLA, Germany

SFR 31: NATAN, Germany

SFR 32: SOLTEC, Germany

SFR 33: 500 kW sodium loop, India

SFR 34: BIM, India

SFR 35: INSOT creep loop, India

SFR 36: INSOT fatigue loop, India

SFR 37: LCTMF, India

SFR 38: LCTR, India

SFR 39: LEENA, India

SFR 40: SADHANA, India

SFR 41: SAMRAT, India

SFR 42: SGTF, India

SFR 43: SILVERINA sodium loop, India

SFR 44: SOWART, India

SFR 45: Subassembly hydraulic test rig, India

SFR 46: TSTF, India

SFR 47: AtheNa, Japan

SFR 48: CCTL, Japan

SFR 49: MELT, Japan

SFR 50: PLANDTL, Japan

SFR 51: SAPFIRE, Japan

SFR 52: SWAT, Japan

SFR 53: ITSL, Republic of Korea

SFR 54: SOFUS, Republic of Korea

SFR 55: STELLA-1, Republic of Korea

SFR 56: STELLA-2, Republic of Korea

SFR 57: RIGADYN, Latvia

SFR 58: TESLA, Latvia

SFR 59: ST-300, Latvia

SFR 60: 6B, Russian Federation

SFR 61: AR-1, Russian Federation

SFR 62: B-2, Russian Federation

SFR 63: BFS1, Russian Federation

SFR 64: BFS2, Russian Federation

SFR 65: Pluton, Russian Federation

SFR 66: Protva-1, Russian Federation

SFR 67: SGDI, Russian Federation

SFR 68: SGI, Russian Federation

SFR 69: SPRUT, Russian Federation

SFR 70: V-200, Russian Federation

SFR 71: ALEX, United States of America

SFR 72: Creep lab–MPM, United States of America

SFR 73: Fracture mechanics laboratory, United States of America

SFR 74: Hydraulic lab–MPM, United States of America

SFR 75: MCF, United States of America

SFR 76: Material processing facility, United States of America

SFR 77: METL, United States of America

SFR 78: SNAKE, United States of America

SFR 79: UW sodium loops 1 and 2, United States of America

LFR 1: COMPLOT, Belgium

LFR 2: CRAFT V2, Belgium

LFR 3: E-SCAPE, Belgium

LFR 4: Helois III, Belgium

LFR 5: HLM, Belgium

LFR 6: Liliputter, Belgium

LFR 7: LIMETS1, Belgium

LFR 8: LIMETS2, Belgium

LFR 9: LIMETS3, Belgium

LFR 10: LIMETS4, Belgium

LFR 11: MEXICO, Belgium

LFR 12: MYRRHABELLE, Belgium

LFR 13: SHAKESPEARE, Belgium

LFR 14: RHAPTER, Belgium

LFR 15: VENUS-F, Belgium

LFR 16: CLEAR-0, China

LFR 17: CLEAR-S, China

LFR 18: KYLIN II M, China

LFR 19: KYLIN II S, China

LFR 20: KYLIN II TH FC, China

LFR 21: KYLIN II TH MC, China

LFR 22: KYLIN II TH NC, China

LFR 23: CALLISTO, Czech Republic

LFR 24: COLONRI I, Czech Republic

LFR 25: COLONRI II, Czech Republic

LFR 26: MATLOO Pb, Czech Republic

LFR 27: HLM EF, European Union

LFR 28: BACCARA, France

LFR 29: MASURCA, France

LFR 30: CORELLA, Germany

LFR 31: CORRIDA, Germany

LFR 32: COSTA, Germany

LFR 33: CRISLA, Germany

LFR 34: ELEFANT, Germany

LFR 35: FRETHME, Germany

LFR 36: MINIPOT, Germany

LFR 37: nELBE, Germany

LFR 38: TELEMAT, Germany

LFR 39: THEADES, Germany

LFR 40: THESYS, Germany

LFR 41: CIRCE–HERO, Italy

LFR 42: CIRCE–SGTR, Italy

LFR 43: GIORDI, Italy

LFR 44: HELENA, Italy

LFR 45: LECOR, Italy

LFR 46: LIFUS5, Italy

LFR 47: NACIE-UP, Italy

LFR 48: RACHELE, Italy

LFR 49: SIRIO, Italy

LFR 50: SOLIDX, Italy

LFR 51: Test stand for fundamental LBE technology, Japan

LFR 52: High temperature material corrosion test loop, Japan

LFR 53: JLBL-3, Japan

LFR 54: JLBL-4, Japan

LFR 55: TEF mock-up, Japan

LFR 56: TEF, Japan

LFR 57: HELIOS, Republic of Korea

LFR 58: IPUL, Latvia

LFR 59: B-2, Russian Federation

LFR 60: BFS-1, Russian Federation

LFR 61: BFS-2, Russian Federation

LFR 62: SGDI, Russian Federation

LFR 63: SGI, Russian Federation

LFR 64: SPRUT, Russian Federation

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18-03391

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978–92–0–101018–6
ISSN 1995–7807