



Status of Molten Salt Reactor Technology

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Status of Molten Salt Reactor Technology

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FOREWORD

The IAEA supports Member States in the area of advanced reactor technology development by providing a major fulcrum for information exchange and collaborative research programmes. The IAEA's activities in this field are mainly carried out within the framework of several work areas, typically supported by Technical Working Groups which assist in the implementation of corresponding IAEA support, and ensure that all technical activities are in line with expressed needs of the Member States. The current advanced reactor technologies supported in this way are advanced light-water reactors, fast reactors and gas-cooled reactors. Sub-programmes are also dedicated to small modular reactors and non-electric applications. No specific program currently exists on molten-salt reactors but the activities undertaken are based on the 60th General Conference (GC) resolution related to molten salt reactors included in the section on the Strengthening the Agency's activities related to nuclear science, technology and applications: "Encourages the Secretariat to explore, in consultation with interested Member States, the need for closer collaboration in technology development for advanced reactor lines by hosting a workshop with the aim of considering launching a new project on molten salt and molten salt cooled advanced reactors;"

The IAEA GCs 61 and 62 also reflect the interest of Member States in molten salt reactors. In particular, GC 61 stated "(n) Welcoming the increased participation at the meeting, organized in November 2015, to "present and share important information on the interest and status of technology developments in the area of molten-salt and molten-salt cooled advanced reactors" and welcoming the meeting that took place in November 2016, ..." GC 62 also mentioned "(o) Noting the increased interest in technology developments in the area of molten salt and molten-salt cooled advanced reactors, ..." Both GCs concluded "20. Recommends that the Secretariat continue to explore, in consultation with interested Member States, activities in the areas of innovative nuclear technologies, such as ... Generation IV nuclear energy systems including ... molten salt nuclear reactors, with a view to strengthening infrastructure, safety and security, fostering science, technology, engineering and capacity building via the utilization of existing and planned experimental facilities and material test reactors, and with a view to strengthening the efforts aimed at creating an adequate and harmonized regulatory framework so as to facilitate the licensing, construction and operation of these innovative reactors;..."

This publication was prepared in response to this interest and summarises the current knowledge on the status of research, technological developments, reactor designs and experiments in the area of advanced reactors that are fuelled or cooled by a molten salt. It is targeted at government officials with technical background from Member States, research institutes, and university students working on related topics, commercial organizations working or planning to design and build MSR and IAEA internal staff, such as the Departments of Nuclear Energy, Nuclear Safety and Security, and Technical Cooperation. It will specifically benefit Member States that are interested in deploying nuclear power for the first time and that wish to understand the technology and needs of research and development for molten salt reactors, which may be used for electricity and cogeneration applications. It aims to present a balanced view not only on the status and potential advantages of the technology of molten salt reactors, but also to identify challenges and technology areas where research and development are required before deployment is achievable.

The IAEA technical officers responsible for this publication were G. Martinez-Guridi and F. Reitsma of the Division of Nuclear Power of the Department of Nuclear Energy.

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1. INTRODUCTION

1.1. BACKGROUND

The International Atomic Energy Agency (IAEA) fosters international exchange of information on the advances in nuclear reactor technology, and supports Member States by providing objective and reliable information and understanding of various reactor technologies. In particular, the IAEA's Nuclear Power Technology Development Section (NPTDS) promotes international collaborative research and development (R&D) in the area of advanced reactor technologies that are needed to meet increasing energy demands. This includes Molten Salt Reactors (MSRs), small and medium sized reactors, and non-electric applications of nuclear power.

There is growing interest in Member States on MSR technologies, and an increasing number of developments and deployment activities in the near term are reported. The overall NPTDS' objective regarding these technologies is to enhance the prospects for the future demonstration and implementation of MSRs.

Over the last few years, NPTDS staff organized several meetings on MSR technology. A Consultancy Meeting in 18–20 November 2015 concluded that there were no known fundamental technical hurdles that would prevent MSR technology from being a safe and feasible energy solution, and that this technology could be commercially demonstrated within the next two decades, provided the necessary funding is available. However, many engineering challenges have to be solved, and the economic competitiveness needs to be studied further. Subsequently, and to fulfil a resolution of the IAEA's 60th General Conference (GC), a Technical Meeting on the Status of Molten Salt Reactor Technology was held from 31 October to 3 November 2016. The need to prepare a publication documenting MSR technology was recognized at this meeting. Afterwards, Consultancy Meetings for developing this publication took place 25–27 September 2017, 29 July to 1 August 2019, and 21–23 July 2020.

1.2. OBJECTIVE

This Technical Report Series (TRS) summarises the current knowledge on the status of research, technological developments, reactor designs and experiments in the area of advanced reactors that are related to MSRs. For the purpose of this document, an MSR is any reactor where a molten salt has a prominent role in the reactor core (i.e., fuel, coolant, and/or moderator).

The publication is targeted at government officials with technical background from Member States, research institutes, and university students working on related topics, commercial organizations working or planning to design and build MSRs and IAEA internal staff, such as the Departments of Nuclear Energy, Nuclear Safety and Security, and Technical Cooperation. The publication will specifically benefit newcomer Member States that wish to understand the technology and R&D needs of MSRs for electricity and cogeneration applications.

The publication aims to present a balanced view not only on the status and potential advantages of the MSR technology, but also to identify challenges and technology areas where research and development are required before deployment is achievable. The work for preparing this publication, including the technical and consultancy meetings mentioned above, provided Member States with the opportunity to share information on their MSR programmes and

projects. This should influence and give directions for possible future collaborative efforts under the IAEA's auspices.

1.3. SCOPE

This publication is intended to document the current status of technology of MSR in Member States, including a short history of technology development and a classification of MSRs. This technology has been evolving and keeps changing with new developments and concepts appearing over time. In some cases, the information reported in this publication was limited due to proprietary restrictions.

1.4. STRUCTURE

The publication is structured in eight sections. Section 2 gives a brief history of MSR technology. Section 3 discusses advantages and technological challenges of MSRs. Section 4 offers a classification (taxonomy) of MSR families (types of MSRs), including a description of each MSR family. Section 5 elaborates on current R&D activities on MSR technology. Section 6 discusses current challenges for deploying MSRs, and finally, Section 7 provides the summary and conclusions.

The publication also offers 5 appendices. Appendices I through IV provide details about the history of MSR technology in Poland and Switzerland, China, France, and Russia, respectively. Appendix V describes MSR concepts, including those that may still be at a conceptual level;¹ it also briefly shows the way in which each concept fits into one of the families of Section 4 by presenting the main characteristics of the concept.

¹ A more detailed description of MSR concepts can be found in the online IAEA's database Advanced Reactor Information System (ARIS) (if the developer of the concept provided such description to this database).

2. HISTORY OF MSR TECHNOLOGY

2.1. US DEVELOPMENT EFFORTS

The technology for molten salt reactors (MSRs) arose from multiple sources. Much of the salt handling and measurement technology was originally developed for the chemical processing and aluminium smelting industries in the early part of the 20th century. The corrosion resistant alloy technology employed for the piping and containers derives from the alloys for high temperature originally developed for gas turbines and jet engines. The concept of employing a liquid-slurry mixture of fuel and moderator dates from shortly after the discovery of fission when Halban and Kowarski at the Cavendish Laboratory in England performed experiments with a slurry of uranium oxide (U_3O_8) in heavy water [1]. Heterogeneous reactors² using solid fuel were, however, selected as the primary path for the Manhattan project because neither enriched uranium nor deuterium were available. Interest in reactors with a fluid fuel was revived in 1943 when larger quantities of heavy water became available with the group led by H. C. Urey at Columbia University, which was investigating slurry reactors using U_3O_8 and D_2O . Heterogeneous configurations of slurry reactors then became the focus of attention as reactor theory indicated that providing a region not containing fuel for neutrons to slow down while avoiding resonance capture would maximize the neutron multiplication. Various slurry designs were pursued during World War II as backups to the Hanford reactors for producing plutonium. Once the Hanford reactors became operational, interest in alternate plutonium production reactors diminished, and nearly all developmental research had been discontinued by the end of 1944.

Scientific investigation of reactors using a fluid fuel, however, continued at both Los Alamos National Laboratory (LANL) and Oak Ridge National Laboratory (ORNL) with objectives to provide power for remote locations and to produce radioisotopes. The physicists were also interested in this type of reactor as a research facility to produce high neutron fluxes. The possibility of the ^{232}Th - ^{233}U breeding cycle in a homogeneous reactor was described in 1944 by L. W. Nordheim [2]. Work on a Th-U aqueous homogeneous reactor with a three-year doubling time continued through 1945. However, technical difficulties with the radiolytic decomposition of water resulting in bubbling reactor instability, lack of container materials with high strength and low absorption of neutrons to enable high pressure (reducing bubbling) operation, corrosion, solution stability, and the explosive potential of the hydrogen bubbles led to a temporary cessation of design and development activities of breeder reactors with fluid fuel. Experiments to establish the feasibility of molten salt fuels were begun in 1947 on the initiative of V. P. Calkins, K. Anderson, and E. S. Bettis [3].

In early 1949, A. M. Weinberg recommended reconsidering reactors with fluid fuel in light of the technical developments since the end of 1945. By July of 1949, a development effort on homogeneous reactors was underway at ORNL. In September of 1949, ORNL was designated by the US Atomic Energy Commission (AEC) as the lead for reactor research for aircraft nuclear propulsion (ANP), which bifurcated the development effort for reactors with fluid fuel into an aqueous branch at low temperature and a salt branch at high temperature. Caustic soda (NaOH) was initially a leading candidate carrier salt for high temperature as fluoride salts are not sufficiently self-moderating to enable a homogeneous configuration. However, problems

² See Section 4 for a brief description of homogeneous and heterogeneous reactors.

with corrosion, the limited solubility of uranium, and the almost complete lack of thorium solubility in NaOH resulted in focusing on fluoride salts by the summer of 1950 [4].

R. C. Briant of ORNL suggested the use of the molten mixture of UF_4 and ThF_4 , together with alkali metal fluorides, as the fluid fuel [5]. Fast reactors using chloride salt were also considered in early stages [6, 7], but the chlorine-35's relatively high neutron capture cross section (and the lack of available technology for chlorine isotope separation at large scale) led to the focus on fluoride salts [8]. Additionally, reactors with fast spectra would require power densities that are very high, necessitating unproven heat transfer technologies, to avoid excessive fissile inventories. Nevertheless, the fused salt fast breeder reactor based on fluoride salts and the Th-U cycle [9] was proposed by a group of students at Oak Ridge School Of Reactor Technology (ORSORT) in 1953. It is a predecessor of the molten salt fast reactor (MSFR) design, which is presented in Section 4. Since it was based on LiF-BeF_2 carrier salt, the performance, especially the doubling time, was worse than for the case of a reactor moderated by graphite.

The ANP program grew rapidly in the early 1950s and many of the technologies of current MSR designs have their origins in this period. The first MSR, the Aircraft Reactor Experiment (ARE), was built and operated at ORNL in 1954. Basic understanding of molten salt production [10] as well as the container materials [11] and components [12] had to be established prior to operating the reactor. Research on salt phase equilibria [13, 14], salt purification methods [15], and corrosion chemistry [16], was carried out during this period. As the ARE was not intended to operate for an extended period, much of the remainder of the aircraft reactor program was devoted to developing technologies to extend the period of operation. Development of a Ni-Mo alloy container material having the property of low corrosion (INOR-8, now UNS 10003 or Alloy N) was among the most significant MSR technical developments of the latter half of the 1950s [17]. The technical accomplishments of the US MSR program through the mid-1950s were summarized in the book *Fluid Fuel Reactors* [1] produced by the AEC for the second United Nations (UN) Atoms for Peace Conference in 1958.

The aircraft reactor program wound down as impractical at the end of the 1950s. The emphasis of the US fluid fuelled reactor program transitioned to generating power for the civilian grid in the late 1950s. In 1956, H. G. MacPherson formed a group to study the performance and characteristics of converter and breeder MSRs [8]. In the mid-1950s, the US was pursuing three different reactor classes using liquid fuel (aqueous homogeneous, liquid salt fuel, and liquid metal fuel). The AEC lacked the resources to pursue all three, so commissioned a task force to evaluate which (if any) of the reactor classes to pursue [18]. The report of the task force for reactors using fluid fuel began with the statement "The molten salt reactor has the highest probability of achieving technical feasibility." The report also noted that maintenance is the most important factor influencing the practicability of MSRs. Substantial efforts were made in the late 1950s to demonstrate that remote maintenance of highly radioactive components was possible using long handled tooling operated from overhead cranes. By the late 1950s, adequate progress had been made in all areas of MSR technology to proceed with design and construction of an experimental reactor to "demonstrate the safety, dependability, and serviceability of a molten-salt reactor and to obtain additional information about graphite in an operating power reactor" [19].

Design of the Molten Salt Reactor Experiment (MSRE) was begun in 1960, construction was initiated in 1962, and initial criticality achieved in 1965 [20]. The MSRE reactor fuel mixture nominally consisted of 65 ^7LiF , 29.1 BeF_2 , 5 ZrF_4 , and 0.9 UF_4 (mole percent). Unclad, fine grained graphite served as the MSRE's moderator. All of the other salt wetted components were fabricated from Alloy N. MSRE reached its full power (7.34 MW) in 1966. Operation

with ^{235}U (~32% enrichment) continued until 1968 when this uranium isotope was removed from the salt and replaced with ^{233}U . Operations with the latter isotope (~91.5% [21]) continued until December of 1969. The last few refuelling additions were performed using $^{239}\text{PuF}_3$. Overall, operation of the MSRE was highly successful, and essentially no problems were encountered with the primary system during operation.

However, four significant technology issues arose during the late 1960s that impacted the design of future MSR:

- (a) Neutron embrittlement of nickel-based alloys at high temperature;
- (b) Radiation damage and dimensional changes to graphite at high fast neutron fluences;
- (c) Liquid-liquid chemical extraction processes for removing protactinium and uranium from molten salt fuel were needed;
- (d) Rapid tritium permeation of alloys that are tolerant to salt at high temperature

In 1972, following MSRE's successful operation and prior to initiating a larger technology demonstration program, the AEC performed an evaluation of MSR technology, documented in WASH-1222 [22]. In addition to the technology issues mentioned above, WASH-1222 also indicated that engineering development of large components, better understanding of the behaviour of fission products, and adequate remote inspection and maintenance techniques would be necessary before MSRs would be suitable for engineering development. WASH-1222 also cautioned that independent of technology capabilities, MSRs were not adequately high on the US government's development priority ranking to have reasonable assurance of the needed sustained resource allocation. It stated:

“When significant evidence is available that demonstrates realistic solutions are practical, a further assessment could then be made as to the advisability of advancing into the detailed design and engineering phase of the development process including that of industrial involvement. Proceeding with this next step would also be contingent upon obtaining a firm demonstration of interest and commitment to the concept by the power industry and the utilities and reasonable assurances that large scale government and Industrial resources can be made available on a continuing basis to this program in light of other commitments to the commercial nuclear power program and higher priority energy development efforts.”

Much of the US MSR development work during the 1970s focused on addressing the identified MSR technology issues. Two tracks were pursued to alleviate the containment alloy embrittlement vulnerability – shielding and improved alloy design. The shielding approach added a roughly half meter interior graphite lining to the interior of the reactor vessel to prevent a significant neutron flux from reaching the vessel wall. The improved alloy design approach was based upon creating large numbers of helium traps (finely dispersed carbides) within Alloy N's microstructure to prevent generated helium from migrating to the grain boundaries. Niobium modified Alloy N, which exhibited improved resistance to neutron embrittlement up to 650 °C, was a key MSR technology advancement of the 1970s [23]. Accommodating the radiation damage characteristics of graphite was also approached both through reactor design and graphite technology development. MSR designs lowered the power density in the core and hence the rate of radiation damage to graphite components. The MSR design with two fluids was also abandoned in large part because of the requirement for graphite plumbing to be in a region of high flux. Development of graphite with acceptable tolerance to radiation damage was a key focus of the gas reactor program during the 1960-70s. While moderator materials for high temperature and increased tolerance to this damage would enable MSR cores with higher

power density, adequate nuclear grade graphite (largely derived from the high temperature gas cooled reactor program) was developed in the 1960-70s. A key technology requirement of the MSR with thermal spectrum and a single fluid was to demonstrate the chemical steps in a liquid-liquid extraction process for removing protactinium and uranium from molten fluoride salts [24, 25]. Calculations indicated that the process could be carried out rapidly and continuously, and that the process equipment would be relatively small [3]. Engineering development studies on processing of fuel salt continued through the mid-1970s [26]. Tritium can be a radiation hazard, and the problem of its escape into the environment was addressed by employing a loop with coolant salt that chemically trapped the tritium before it could reach the steam cycle. Technology for tritium trapping was demonstrated at engineering scale in the mid-1970s [27]. The US MSR program also began the process of large scale, molten salt, hydraulic components. Thermal and hydraulic design studies for a nuclear qualified steam generator for a large MSR were completed by an industrial designer in the mid-1970s [28].

The issue of proliferation vulnerability of nuclear fuel cycles also arose during the 1970s. US MSR designs that existed in 1976 were not focused on making diversion of fissile material difficult and easily detectable [29]. President Carter's nuclear power policy statement of April 7, 1977 announced "we will direct funding of U.S. nuclear research and development programs to accelerate our research into alternative nuclear fuel cycles which do not involve direct access to materials that can be used for nuclear weapons" [30]. A conceptual design for an MSR that avoided direct access to materials that could be used for nuclear weapons was subsequently developed in the late 1970s [31]. Overall, the US MSR program largely overcame the technical issues identified in WASH-1222 as being necessary to resolve prior to entering into engineering development and developed designs that were compliant with the proliferation resistance policy directive. However, the program never became of sufficiently high importance to obtain the required resources. US MSR activities at large scale ended by 1977.

2.2. NON-US DEVELOPMENT EFFORTS

Several Member States began efforts for evaluating MSR technology following the reporting of US activities at the second UN Atoms for Peace conference in 1958.

The MSR research in Switzerland started in the late 1960s when Mieczyslaw Taube joined the Swiss Federal Institute for Reactor Research (EIR), the predecessor of the Paul Scherrer Institut (PSI). He is one of the pioneers in the chloride salt reactor research. He published his first paper related to a chloride fast MSR [32] at the Department of Radiochemistry of the Institute of Nuclear Research in Warszawa in 1961, and in 1967 he proposed the concept of cooling by boiling aluminium trichloride, which was in direct contact with the fuel salt [33]. After he joined EIR, the boiling aluminium trichloride was assessed as a coolant of a fast reactor with solid fuel [34]. The later research in Switzerland was mainly oriented on heterogeneous chloride fast MSRs [35], where the blanket salt was often used as a coolant for the fuel salt. At that time, when the ORNL MSR project was suspended, three labs of EIR were partly involved in MSR neutronics, chemistry, and materials research. The research at EIR was not stopped; however, the focus moved to homogeneous fast chloride MSRs, including combined breeding in Th-U and U-Pu fuel cycles [36] from the middle 1970s. At the end of the 1970s, Taube and his team proposed the SOFT (Salt reactor, On site reprocessing, Fast converter, Task) reactor [37] as the last concept before the MSR research was suspended at EIR. The thesis of Ottewitte [38], who was involved with EIR in the 1970s, represents the continuation of this work to some extent. Appendix I provides further information on the MSR history in Poland and Switzerland.

The MOSEL reactor concept was studied in the Federal Republic of Germany in the mid-1960s [39]. This breeder concept employed a salt bearing thorium as the primary coolant fluid with the fuel salt in tubes or plates within the core connected to inlet and outlet manifolds and circulated primarily for the purposes of fuel mixing, addition, and removal [40].

UK efforts to develop nuclear graphite that is tolerant to radiation damage for their gas cooled reactors contributed significantly to MSR. UK work concentrated on reactors using a chloride salt, a fast neutron spectrum, and the ^{238}U -Pu fuel cycle. UK work on chemistry of chloride fuel salts began in 1965 and was summarized in 1969 [41]. The UK Atomic Energy Authority examined three variants of a 2500 MW(e) molten chloride fast reactor 1) in-core direct contact with lead, 2) in-core cooling by passing a blanket salt through molybdenum tubes, and 3) out-of-core cooling [42].

The Netherlands began MSR research in 1963 as a joint undertaking between Euratom Joint Research Center (Petten Establishment) and Delft University of Technology (TU Delft). Due to budgetary restrictions at ORNL, staff from the laboratory of Thermal Power Engineering at TU Delft were recruited to investigate molten salt steam generators with bayonet tubes. These efforts were superseded by the Delft Molten Salt Project (DMSP), lasting from 1969 until 1978. This project mainly focused on component and engineering developments of primary fluoride and secondary nitrate molten salt loops for the scale of a pilot plant [43, 44, 45]. Valves, pumps, heat exchangers, steam generators, and flow meters were investigated, along with simulation software capable of predicting operating limits imposed by steam generators of various geometries [43].

In February 1970, Premier Zhou Enlai's stated that Shanghai should develop a nuclear power plant (NPP) to solve the disparity between Shanghai's economic development and energy shortage. An MSR for high temperature was initially selected for the NPP project, which would generate 25,000 KW(th) [46]. In the 1970s, a Critical Experiment Device (hereafter the device) was established for the research on the physics characteristics of MSR in the Shanghai Institute of Applied Physics (SINAP) (then Shanghai Institute of Nuclear Research), and a series of zero power experiments and related results were obtained. The device was later transformed into a Uranium-Water Lattice Experiment Device for changing the research orientation. A major advantage of this device is the flexibility of the fuel elements and moderator elements arrangement in the core allowing different moderating ratios. The purpose of the device was to experimentally verify the results from theoretical calculations; get important characteristics, such as the static and dynamic features, the unit inventory gram reactivity, and the temperature effect of the molten salt used; and obtains the relevant design data such as the control rod calibration and its temperature effect, and the fertile fuel conversion ratio [47, 48, 49, 50]. Appendix II offers additional information on the MSR history in China.

Activities on MSR technology in Japan began in the late 1970s [51, 52], focused on mostly small-sized FUJI designs having self-sustaining capability, and were proposed for power generation. In addition, an off-line reprocessing facility and an accelerator driven system (ADS) for producing fissile material were proposed.

The history of MSR technology development in France and Russia are discussed in the next subsections, 2.2.1 and 2.2.2, respectively.

2.2.1. History of development efforts in France

In France, the Commissariat à l'Énergie Atomique and Electricité de France pursued MSR from 1973-1983 [53]. The French work involved four experimental loops, including development of a pump and a heat exchanger that were made from graphite. Direct contact, molten lead cooling was evaluated to minimize fissile inventory while avoiding metallic materials in the core.

After the end of Super Phénix Sodium Fast Breeder Reactor (December 1998), and in the frame of the 1991 French law on wastes [54], innovative solutions were studied in order to burn the plutonium and minor actinides (MA: Americium, Curium, Neptunium) coming from the French PWRs. Among different reactors, Molten Salt Reactors were identified as one of the most promising by many papers [55, 56, 57, 58]. While those studies considered subcritical systems, driven by a particle accelerator, Electricité De France (EDF) proposed a critical burner for plutonium and minor actinides called AMSTER (Actinide Molten Salt TransmutER) for the sake of simplification.

The French R&D on molten salt reactors had two phases. The first phase from 1998 to 2005 was an in-depth optimization and study of a reactor of the type Molten Salt Breeder Reactor (MSBR). AMSTER was defined by EDF and the TMSR by the French National Centre for Scientific Research (CNRS). The first phase concluded that no design of MSBR-like reactor moderated by graphite allows to reach all the design goals, that is, iso-breeding, negative reactivity feedback coefficients, and reasonable graphite lifetime. The decision was taken to avoid using graphite and develop non-moderated MSR concepts.

Accordingly, the second phase of French R&D from 2004 to the time of publication of this document involved these concepts. The CNRS proposed the non-moderated Thorium Molten Salt Reactor (TMSR) [59, 60, 61] in 2004, which uses a fluoride salt and is iso-breeder. This concept was renamed MSFR and is described in Subsection 5.7 on French R&D activities. EDF also proposed a non-moderated iso-breeder MSR in 2004, the REBUS-3700 [62, 63] with a classical depleted U / Pu cycle. This reactor uses a chloride salt (38UCl₃-7TRUCl₃-55NaCl) because a preliminary study showed that ²³⁸U / ²³⁹Pu breeding is difficult to reach with a fluoride salt, since the neutron moderation by fluoride is too high. Both non-moderated reactors show largely negative reactivity feedback coefficients and are iso-breeder with a limited fuel reprocessing (around 40 l of salt reprocessed per day).

Appendix III provides detailed technical information about the French MSR designs developed from 1998 to 2008.

2.2.2. History of development efforts in Russia

The Soviet MSR program began in 1976 [64, 65], and was broadly based, including the salt-cooled reactors [66], thermal spectrum MSRs, and eventually the fast spectrum systems. It notably included an in-core, natural circulation liquid-salt fuelled loop [67].

The Kurchatov Institute (KI) was the main organization under which various specialized institutions collaborated. A reduction of activity and general stagnation of the nuclear industry happened after the Chernobyl accident in 1986. However, there was an increase of conceptual studies as a result of the interest in inherently safe reactors of a new generation by the end of the 1980s. These studies of MSR technology were mainly directed on developing thorium–uranium concepts.

The MSR program addressed the following topics:

- reactor physics and reactor safety;
- container materials for fuel and coolant salts;
- physical and chemical properties of molten salt mixtures;
- heat transfer and hydraulics of fuel and coolant salts;
- handling and circulation of fuel and coolant salts;
- process and radiochemical bench tests of model installations;
- radiochemistry of molten fluoride fuel salt.

The first topic only included theoretical studies, and the rest involved both theoretical and experimental ones. Some of these topics are described in Appendix IV.

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3. ADVANTAGES AND TECHNICAL CHALLENGES OF MSR TECHNOLOGY

3.1. ADVANTAGES OF MSR TECHNOLOGY

Significant research on MSR technology is underway in various Member States. Early development work conducted as part of the Molten Salt Reactor Program (MSRP) at the ORNL beginning in the late 1940s is the origin of most of the work currently being carried out on a variety of MSR concepts. As most of these concepts use fuel in liquid form (the molten salt contains the fuel and is the coolant) or solid fuel with cooling by molten salt, a very different set of features characterizes this class of nuclear reactor when compared to traditional water cooled reactors (WCRs).

Below is reported an overview of significant advantages of MSRs that use a molten salt to contain the fuel and as the coolant over conventional WCRs. Overall, some of the advantages discussed are also applicable to MSRs with solid fuel and cooled by a molten salt. Roughly all the advantages listed hereafter can be divided into two overarching typologies, namely safety and economics; these are described in subsections 3.1.1 and 3.1.2, respectively. Generally, MSRs also have potential advantages from the points of view of environmental benefits and non-proliferation of nuclear material, which are reported for simplicity among the safety advantages in Subsection 3.1.1. Some features of MSR technologies are beneficial to both safety and economics; for example, operating the primary system at near atmospheric pressure benefits both. Subsection 3.2 discusses the technical challenges to deploying MSRs and R&D needed to address them.

3.1.1. Safety advantages

Near-atmospheric operating pressure. The possibility to operate the reactor at near atmospheric pressure brings several advantages in terms of operation, safety and economics. The reduction of the mechanical stresses to the structural materials entails a lower risk due to potential accident scenarios, as well as a reduction of the costs, compared to the typical WCR. In addition, a salt circuit and other confinement barriers at ambient pressure reduce the possibility of radioactive nuclide dispersion in case of such scenarios.

Radioisotope retention. Several benefits are gained from the chemical and thermophysical properties of the molten salts. Molten salts are exceptional solvents and strongly bond many of the fission products, even at high temperatures. This is a major, intrinsic safety feature, as the salt (both in liquid and in solid fuel systems) becomes an additional barrier to the release of radioisotopes to the atmosphere. Conversely, WCRs generate a much wider range of unretained radioisotopes, which accumulates in the fuel pins increasing their internal pressure. The volatile compounds in the fuel pins are highly mobile and become the main source of radioactive contamination in the event of an accident.

Chemically inert and not flammable. The molten salt has low chemical reactivity. Not reacting exothermically with nearby materials, air, or water is a significant advantage. There is no potential for hydrogen explosions or sodium fires. Overall, containment of radioisotopes in MSRs represent an easier task when compared to nuclear reactors using high pressure fluids (such as water or helium) or reactive fluids such as sodium [68].

Removal of gaseous fission products. Since gaseous fission products can be removed by separating them from the fuel salt during reactor operation, pressure build-up within the core

can be avoided, mitigating both safety risks of the reactor compartment and the consequences of accident scenarios involving the core.

No irradiation damage or mechanical failure of fuel. Molten salt is an ionic liquid, and irradiation damage or mechanical failure of the fuel does not occur.

Strong negative reactivity temperature feedback. This feedback is predominantly due to the effects of nuclear Doppler, fuel salt expansion, moderator temperature or a combination of these effects resulting in a strong negative reactivity temperature coefficient that characterizes most MSR concepts using liquid fuel and providing an intrinsic stability of the core and passive shutdown for any temperature rise. It enhances nuclear safety and other benefits like load following. Some early designs moderated by graphite, such as the 1970s ORNL's MSBR, did feature only a weakly negative overall temperature reactivity coefficient due to a positive density term and a positive graphite temperature term only just compensated by a strong negative Doppler term. These issues are being overcome in modern efforts. The density term can be made negative by utilizing a larger fraction of graphite, resulting in a higher carbon to fissile ratio. The positive graphite term of the MSBR was due to using ^{233}U ; however, the ^{235}U gives a beneficial negative graphite temperature reactivity term for MSRs on burner cycles using low enriched uranium (LEU). Thus, MSR designs with fast and thermal spectra can achieve strong total negative reactivity feedback, as well as all individual terms negative.

Molten salt as a heat sink. The molten salt will absorb some of the decay heat after reactor shutdown. The total heat sink of the core itself can be very high in the case of moderated designs due to the large mass of moderator, typically graphite. In fast and thermal MSRs with liquid fuel, there is a tendency for higher thermal inertia than their respective counterparts with solid fuel. More in general, for most MSR designs the liquid form of the fuel provides a simpler way to remove heat by circulating the fuel in a heat exchanger. Moreover, heat removal does not have to occur in the reactor core as for WCRs, eliminating the risk of circulating coolant fluid in the core.

High boiling point. Molten salts are usually characterized by a high boiling point. This physical property, in combination with the salts' heat capacity, allows a large margin (up to 700 °C) between the operational and the boiling temperatures to accommodate temperature transients in case of potential accident scenarios without losing the salts' cooling capability.

Excellent neutron economy. MSR technology allows for better utilization of fresh fuel loaded in the core than some of the other reactor technologies. This is mainly due to the absence of metallic structure, burnable poisons or other parasitic neutron capture elements within the core. The addition also of semi-continuous fission product removal processes can further improve neutron economy to allow break even or breeding operation.

Flexible fuel cycle. A reactor core with liquid fuel can host all actinide elements and enable their continuous recycling. Since the fuel manufacturing stage can be omitted, even elements producing intensive decay heat can be hosted and recycled with the liquid fuel. Solid fuel, on the contrary, cannot be chemically adjusted following manufacturing. They also are vulnerable to mechanical and radiation damage, and hence must be checked and replaced periodically.

The liquid fuel cycle flexibility supports both the breeding of fissile isotopes by using thorium-232 or uranium-238 (enhancing resource utilization), as well as the recycling of transuranic stockpiles (burning existing waste). This is a significant advantage from both environmental

and non-proliferation perspectives. Furthermore, if recycling is done on-site, spent fuel transportation with related risks is diminished.

3.1.2. Economic advantages

The intrinsic safety features of MSR power plants described in the previous subsection should contribute to a simple and safe NPP. Naturally, this has a potentially positive impact on the economics of MSRs, as less engineering would be required to ensure safe operations.

High temperature. MSRs have the possibility to operate at high temperatures ($> 600^{\circ}\text{C}$) with the resulting advantage of very high thermodynamic efficiency (up to 50%). This is possible with either a conventional and standard Steam Rankine cycle, or with the advancing field of Gas Brayton technology, thus improving the economics of the operation of a NPP based on an MSR. In particular, the supercritical CO_2 Brayton cycles offer the potential of dramatically reduced turbine size in the future. The high operating temperatures also support non-electrical applications that require high temperatures. Whereas water cooled technology is limited to the 300°C range, a very large potential market for industrial heat exists for output temperatures around the 550°C range that MSRs can provide. The petrochemical industry, steam electrolysis at high temperature, ammonia and liquid fuel production along with desalination facilities are a few of the potential applications.

Effective load following. Some MSR concepts offer very promising characteristics in terms of operation flexibility, thanks to the liquid fuel. As mentioned above, many MSR concepts using liquid fuel have a negative reactivity temperature coefficient. The negative reactivity feedback acts very rapidly when the heat is produced directly in the coolant, that is, when the fuel salt itself is used as coolant. Some MSRs are thus particularly well suited to load following of the offsite electric grid thanks to their ability to rapidly adjust the power generated to the power extracted, the salt temperature variations remaining very small, as shown by some recent multiphysics calculations of such systems [69]. Indeed, as soon as the fuel salt temperature varies because the power extracted has changed, the quasi-instantaneous variation of the salt density modifies the power generated. Thus, the salt and reactor structure temperature excursions are limited. This property fits well an electricity grid whose energy mix gives a larger share to intermittent sources than a conventional grid. Moreover, an MSR may adjust to the hardly foreseeable needs of the grid without requiring a control rod system. In commercial use, limitations on load following will be due not to reactor physics, but either to limitations in ramp rates of turbine technologies or to limit temperature fluctuations of the reactor. Furthermore, the fact that liquid fuel use allows the passive release of the fission product Xenon results in substantial reduction of the Xenon poisoning that severely limits load following capabilities in reactors with solid fuel and thermal spectrum, such as WCRs.

High resource utilization. In many MSR concepts, the fuel cycle can be closed also for minor actinides [70], unlike in most solid fuel reactors. Accordingly, only reprocessing losses would become a waste, while all actinides from spent fuel of existing reactors can be utilized by MSR breeders or burners in the closed cycle. Even in open cycle, the resources utilization in MSR is higher than in solid fuel reactors. Excellent neutron economy, radiation stability of the carrier salt and the possibility to continuously remove fission products and adjust the fuel composition allow for higher burnup and lower initial fissile fuel enrichment.

Fuel qualification. Whereas fuel qualification of any new solid fuel design requires years of testing to assure the integrity of the solid fuel and its cladding, this fuel qualification process can be far simpler with liquid fuel. While the fuel composition and thermophysical changes

across all stages of an MSR lifetime must be tested, this work can be done without the need of extensive and time consuming irradiation programs, as the fluoride salts (and likely chlorides as well) are not damaged or undergo radiolytic decomposition.

Compact form factor. In liquid fuel designs, the liquid fuel allows optimal utilization of the reactor core volume. For thermal spectrum reactors, this opens to the possibility on NPP modularity and decentralized power production. Modularity and form factor, combined with the load following capabilities mentioned above, make the MSR an attractive technology for the energy mix of modern power grids.

3.2. TECHNICAL CHALLENGES OF MSR TECHNOLOGY

This subsection discusses technological challenges for deploying MSRs. Current R&D efforts for this deployment mainly focus on the areas presented here.

While MSR technology operability was demonstrated in the 1960s during the MSRE program at ORNL, a few technical and developmental challenges lie ahead of modern MSR commercial deployment. These can be divided into three macro areas: (i) reactor physics, (ii) salt chemistry and materials science, and (iii) engineering. While R&D efforts in these areas are often intertwined, the following discussion treats them separately for sake of clarity. Challenges that are of a less technical nature, such as related to the supply chain, licensing, and regulatory aspects are better characterized as non-technical challenges and are addressed in Section 6. Since most challenges are ultimately related to technical issues, there is no clear distinction between technical and non-technical challenges. Therefore, there is some overlap between the challenges discussed in this subsection and those in Section 6.

Performance and fuel cycle requirements have been demonstrated for several MSR designs from a reactor physics standpoint. In general, the physics of MSRs is well understood and there are no fundamental scientific obstacles to the technology. However, some significant hurdles remain. One is the lack of MSR multiphysics simulation software that may be acceptable by a regulatory body. Worrisomely, to build confidence in the software, an experimental verification is required. But to get experimental setups for collecting data, a regulatory approval is needed. Such licensing loophole may be overcome with the joint efforts of MSR developers.

The low energy thermal scattering kernels for various salts is currently not available, nor are displacements per atom (DPA) data for candidates of structural materials that are corrosion resistant. These gaps can be filled only via significant modelling and irradiation experiments. Furthermore, high fidelity simulations are needed to examine a potentially variable circulating source term involving delayed neutron precursors (DNPs), which poses increased requirements on reactor control mechanisms.

The most critical development challenges for MSRs revolve around salt chemistry and materials. Generally, the identification and characterization of suitable salt and material combinations for use in MSRs is an absolute priority. Albeit lengthy and resource intensive, the qualification of new structural materials for advanced reactor applications might become necessary. In this sense, the acceleration of materials discovery and testing via multiscale simulations and machine learning approaches will play a central role [71]. A better definition

of the interface between materials and salt is sought to drive the selection of the appropriate material and salt candidates.

A critical issue concerning the salt phase is the accurate definition of thermophysical and thermochemical properties of fuel and coolant salts. The quality of the existing data is a concern due to lack of quality assurance protocols and standardized experimental methods to generate these data. In general, any MSR vendor would be required to generate data specific for its salt choice, which constitutes a significant effort. A validated salt properties database providing data of fuel and coolant salts as a function of salt mixture composition, temperature, and pressure can be used as a practical and efficient tool for reactor design and modelling of MSR performance. The prediction of these properties via, for example, molecular dynamics simulations, is a second independent tool that can be useful to inform engineering choices during the reactor design phase but will not suffice alone during validation with regulatory bodies.

Thermochemistry data and models that describe the speciation of the fuel, the salt, activation products and fission products will also need development, as they are necessary for performance and safety evaluations.

To mitigate the degradation of structural materials in contact with a molten salt, it is paramount to maintain a strict control of the salt composition, from fuel loading throughout the core lifetime. Compelling challenges here are the definition of manufacturing and purification protocols for the salt compositions, which are needed for a reliable supply chain for the MSR fleet. The purity of the salts must be high to prevent radioactive activation and excessive corrosion. The exact composition of the salts upon loading in the reactor shall be ensured, and hence, storage and transportation protocols are also to be developed, in collaboration with fuel suppliers. As an example, the infrastructure and capacity for separation of lithium and chloride isotopes are currently missing.

Some core components experience exceptionally fast degradation. In moderated (epithermal) MSRs concepts, the choice of graphite as moderator poses certain limitations on the core lifetime or impose relatively fast replacement. Graphite degradation occurs due volume change, structural contraction followed by swelling. The process is dependent on the neutron flux, but more importantly is dependent on temperature, which is distinctively high in MSRs. This determines a relatively short lifetime for the moderator material, and periodic substitution. A significant challenge here lies in the ability to extend the lifetime of graphite materials for moderation, or to implement the use of water without the loss of a good portion of the technological advantages.

During operation, the salt composition has to be maintained within carefully selected limits. A key aspect to mitigate corrosion is the monitoring and control of the salt redox potential, which has been demonstrated in experiments at lab scale, but whose successful implementation in MSR prototypes presents significant challenges. Electrochemical methods to monitor and alter the oxidative power of the salt anions are still to be demonstrated in prototype setups. Electrochemical setups withstanding highly corrosive and irradiated environments must be developed and their functioning over suitable timeframes validated.

In general, instrumentation to sample the salt physicochemical and dynamic properties is already developed for other applications, but must be adapted to use in MSRs, as the harsh conditions pose exceptional stress on instrumentation components. Most of the instrumentation in MSRs will not be in contact with the liquid salt, implying indirect measuring principles to

be used, that are radiation tolerant. Otherwise, salt compatibility must be demonstrated for any instrumentation. In some cases, like MSR with fast spectra, the higher power densities will make in-core instrumentation impractical, increasing the burden on monitoring technique development.

Furthermore, some MSR concepts (e.g. those using the thorium fuel cycle) include an on-line fuel salt reprocessing system, which can increase the fuel circuit complexity significantly. Set aside the security concerns related to fuel inventory and proliferation, the functioning of a chemical process unit must be demonstrated to perform sufficiently well in the treatment of highly radioactive material, separating nuclear waste from purified fuel. While it can be said with confidence that the processing of a liquid fuel form is greatly simplified compared to the use of solid fuels, it can be in no way trivialized the significant development and potential operational costs of on-site processing systems for fuel salt that are proliferation resistant.

Several challenges also concern the process and system engineering of MSRs. In particular, an environment at high temperature and associated issues of thermal stresses on vessels, heat exchangers, and mechanical components is a significant challenge. Fluoride salts are also fluxing agents such that any protective oxide coating of metallic surfaces is absent and makes items such as valves a significant challenge due to self-welding concerns. While early development at ORNL showed promise, much work is still required or methods to avoid mechanical valves are needed (such as the case with the 1960s MSRE). Accurate thermohydraulic studies must be carried out to generate data on fluid mechanics and heat transients. The transport of both heat and mass across the core is set to vary during the fuel cycle, as fission products generate and travel across the fuel circuit. Loop test facilities utilizing full scale components should be developed, to test molten salt circulation and purification at time and size scales relevant to commercialization.

For MSRs, the design of a reliable off-gas system to separate gaseous fission products from the fuel represents a key challenge, albeit some similarities exist with current WCRs. Degassing undeniably represents a convenient feature of liquid fuels; in the past it was also considered in the form of vented pins for some solid fuel reactors. Even though not all MSR concepts rely on it, degassing benefits the neutron economy by removal of the neutron poison isotope ^{135}Xe , easily separated from the melt due to its very low solubility and increases the plant safety as no pressure build-up occurs since this isotope does not accumulate within the core (during normal operations). On the other hand, studies from the MSRE revealed that non-negligible amounts of salt aerosols, solid particulates, and mechanically volatilized metal fluorides compounds are directed outside the primary circuit by degassing operations [72, 73]. Together with other naturally volatile radionuclides (e.g. tritium, and molecular and noble gasses), they form a complex feed for any purification system, which poses greater process design challenges. Established technology, such as filtering, has the obvious drawback of clogging and has limited efficacy; hence developers are devoting resources to find more comprehensive separation and purification methodologies [74, 75]. Finally, depending on the operation philosophy, degassing systems may significantly increase the void fraction present in the primary circuit, causing complex consequences for reactivity control [76].

The liquid fuel form and the removal of some fission products through the degassing or filtering have both consequences to the fuel cycle strategy applicable to MSRs. In liquid fuel, the share of fission products does not need to correspond to the fissioned mass of actinides. From the neutron economy perspective, the achievable burnup in MSRs with degassing can be thus higher than in solid fuel reactors. At the same time, the homogeneous nature of the liquid fuel

causes that there is no difference between average and discharge burnup. In solid fuel reactors, the burnup of discharged assemblies is up to twice higher than the average burnup of the remaining assemblies in the active core. From this perspective, the homogeneous fuel represents a disadvantage. The refuelling operation in solid fuel reactors has two major fuel cycle functions: to replace the fissioned mass of actinides and to reduce the amount of fission products in the active core. Whereas the actinides can be refilled to the homogeneous MSR fuel easily, the removal of the remaining fission products is more complicated. The simplest option is to discharge the fuel salt with average burnup and declare it as a waste in the respective open cycle strategy. A more complex option is to separate the fission products in a reprocessing unit and recycle the actinides and carrier salt back to the reactor in a closed fuel cycle. The frequency of reprocessing can range from small daily removed volumes to entire salt discharge at once after a few years of irradiation. The advantage of frequent reprocessing is the minimal actinides load needed for operation and minimal excess reactivity and salt composition oscillations. The disadvantage is that the irradiated and reprocessed salt have the same average burnup. The advantage of the second extreme option, entire salt discharge at once, is that the discharged burnup is twice higher than the average burnup in the core over time. However, the disadvantage of the latter option is that almost twice higher actinides load is necessary to operate the reactor and the share of fission products in the reactor oscillates. These oscillations can be smaller and thus more acceptable for fast spectrum MSRs. In between of the above-discussed two options there is a full spectrum of reprocessing frequencies. Furthermore, there are options to swiftly recycle fissile actinides from the discharged salt and postpone the separation of fission products and remaining actinides by several years. In general, the salt reprocessing is an additional challenge and the major conceptual decision in this respect is the location of the reprocessing unit. It can be either located at each plant, or centralized at a place outside the MSR plant(s). In the second case, the lower reprocessing frequency will be preferable to minimize the count of radioactive material transport and to prolong the time span for the reprocessing. Nonetheless, the truly online methods of degassing and filtration should be utilized whenever possible to remove part of the fission products. A quasi-batch system, which is intermediate between purely off-line (perhaps on-site) and on-line processing may be most appropriate for reprocessing salt fuel.

Some MSR concepts, typically foreseen as thermal breeders in the closed Th-U cycle, may require ^{233}Pa separation from the salt to improve the neutron economy; see Subsection 4.6, Graphite Moderated MSRs. This separation needs to be done locally and frequently, and it could be an issue from the proliferation and safeguards perspectives.

As for all solid fuel reactors, the high level waste produced by MSR operation strongly depends on the fuel cycle strategy. Actinides can be declared as a waste when discharged from a reactor operated in an open fuel cycle. Alternatively, some of them become a waste as the reprocessing losses in a closed fuel cycle. Hence, the amount of actinides in the waste stream strongly differs depending on the fuel cycle used; however, the difference between solid and liquid fuelled reactors is minimal. Another challenge is the chemical form of the waste. A complex treatment may be needed before the discharged salt can be temporarily or permanently stored. At the same time, there is a possibility that processes using a single step, like for instance vitrification, can be applied to the waste in salt form. Furthermore, due to the degassing and filtration processes, part of the waste stream may require other treatment or storage strategy. In the case of gasses, many small separated storage volumes may be required, so that a single failure does not result in releasing the entire inventory. The final volume of the waste per produced unit of energy is hard to assess because the actinides molar share is very different between the various MSR concepts and the method of waste treatment and immobilization still needs to be assessed.

Similarly, the share of fissile actinides in the waste stream will have a broad range because MSR can be fuelled by enriched uranium, recycled plutonium, or recycled transuranic elements, and an MSR can be developed to be operated as a breeder, converter, or burner.

Depending on the concept, MSRs can also produce substantial volume of middle to high level waste. There are three major sources of radioactive waste in WCRs: production of fission products, transmutation of primordial actinides, and activation of irradiated materials. There is one more major source in MSRs: contamination of materials by actinides or fission products. Whereas in WCRs the water coolant can contain traces of actinides and fission products released from leaking fuel pins, the liquid fuel in MSRs contains a very large amount of actinides and fission products. Accordingly, solid structural materials in direct contact with the salt or the graphite moderator (for a thermal MSR using this kind of moderator) can be strongly contaminated. Since they will be irradiated at the same time, their limited lifespan can result in a substantial stream of middle to high level waste.

The design and manufacturing of certain components, for which little prior experience is available, are among the key challenges that MSRs pose. In particular, great efforts will be required to produce reliable and long-lasting components (such as circulation pumps, heat exchangers, valves, and flanges) that are in contact with the harsh salt system. Suitable instrumentation to monitor the status of these components during their lifetimes is yet to be developed and demonstrated. Remote maintenance is likely required due to the harsh environment, with the consequent rise in the complexity of the operations and cost. Certain MSRs will require components to be periodically replaced, which adds design requirements of accessibility to the core for degraded or spent components to be substituted, increasing the overall costs.

Several accident scenarios must be evaluated before commercialization, independently of the MSR design. To develop accurate source term analysis to gauge performance and safety, developers must describe the chemical speciation and transport behaviour of fission and activation products in the fuel circuit and within the containment area. This analysis has to be tailored for non-equilibrium conditions, which adds to its complexity because radionuclides transport across the plant systems situated within the containment zone changes according to several scenarios, such as normal operation with and without an off-gas system, maintenance, abnormal events, and accident scenarios [77]. To promote the public confidence in the technology, developers can furthermore focus on producing accurate data about the thermodynamics and chemical speciation of fuel salt mixing with the external atmosphere, in the event of loss of coolant accidents. The claims of exceptional solvent ability and inertness of fuel salt should be verified for a variety of spillage scenarios, some general and some that are specific to an MSR design. Spillage of the fuel salt may also occur as part of monitoring, maintenance, or refuelling operations; effective protocols to remediate it have to be demonstrated, albeit the solidification of the salt matrix is expected to simplify the cleaning of contaminated areas. Nuclide inventory considerations are to be addressed too in the event of spillage; in this regard, defining the barriers for preventing release of radioactive material for liquid fuels also is necessary. Spilling fuel salt during any operation is not especially technically challenging provided a catch basin and drain tank (with adequate cooling) are provided. All known designs incorporate this feature. Conceptually, fuel salt flows like water down to the drain tank where it is safely cooled. Some uncertainty, however, remains as to which radionuclides evolve out of the fuel salt during both normal operations and spills.

Finally, among significant challenges to deployment, it is worth mentioning the constitution of standards for various aspects of the materials qualification, such as metals and ceramics fabrication, joining, and weld inspection.

To summarise, the MSRE experiment showed that molten salt reactors can be safely designed and operated. The present challenge is to design and manufacture reactors with sufficient operating lifetime to enable them to be economically viable.

For thermal spectrum MSR, both the fine-grained graphite and Hastelloy-N based nickel alloys necessary to enable construction are currently available, albeit with a limited supply chain. Although these alloys are not included in major high temperature nuclear structural design codes such as ASME BPVC Sec. III, Div. 5 and RCC MRx, as noted above, MSRs are low pressure systems so construction to other codes and standards may be possible. Furthermore, as the reactor vessel is exposed to relatively low neutron fluences, then it is also possible for them to be constructed using a conventional high temperature nuclear material such as 316 stainless steel that is coated or clad to provide the required corrosion resistance. The use of coated or clad materials is more challenging for reactor internals and piping, and hence it is likely that unless new advanced materials are used, these components will have to be routinely replaced since their operational lifetimes will be less than that of the reactor vessel.

The neutron fluence in fast spectrum MSR can be significantly larger and unless sufficient secondary salt shielding is inherent in the design, neutron absorbing materials may have to be also included in the reactor vessels coating/cladding. For the same reason, unless novel, more radiation resistant materials can be developed, the operational lifetimes of reactor internals in fast spectrum MSR will be short, necessitating frequent replacement and maintenance.

4. CLASSIFICATION OF MSR FAMILIES

4.1. INTRODUCTION

Molten salt reactors are a broad and diverse category of reactors, in which a molten salt plays one or more significant functions in the reactor core, such as fuel chemical form, liquid fuel carrier, coolant, or moderator. All historical and recent concepts rely either on fluoride or chloride salts. Other halides or other salts, like for instance the industrially utilized nitrate salt, are not considered because one or more of their properties, such as radiolysis of the salt without recombination, high neutron absorption or activation, low solubility of fuel salt, or unfavourable melting points disqualify them for use in the reactor core. Also, the homogeneous mixture of fluorides and chlorides is not considered. It was evaluated briefly early in the ANP program and recently mentioned by Ref. [78]. Such a mixture would have high neutron absorption in moderated reactors, and softer neutron spectrum than standalone chlorides in fast reactors. Furthermore, it will strongly increase the complexity of the chemistry of the fuel salt.

MSR development spans about seven decades, that is, since the late 1940s. Some of the conceived MSR concepts are based on a relatively simple idea, typically related to the fuel cycle arrangement, and rely only on neutronic simulations; other concepts were supported by enormous research and development (R&D) effort and their Technological Readiness Level (TRL) [79] is very high. A prominent example of a design class that is well developed is the Graphite-based MSRs.

In the past, there were several efforts to classify MSRs, for example, see Refs [42, 80]. The contribution described in Ref. [80] to the International Nuclear Fuel Cycle Evaluation project, which was supported by IAEA, is especially relevant because it includes detailed discussion of an MSR classification. It uses the following classification parameters: method of cooling, flux intensity, number of core zones, type of fuel cycle, neutron spectrum, and salt type. The method of cooling was selected as the foremost criterion and defines three major MSR groups based on the location of the heat exchange between fuel and dedicated coolant, and on whether the coolant and liquid fuel are in direct contact or not. The three groups defined were: internally indirectly cooled MSRs, externally indirectly cooled MSRs, and internally directly cooled MSRs. For internally cooled reactors, the heat exchange between fuel and coolant takes place in the active core; for externally cooled, it is outside of the core. This classification is, however, outdated because 1) the directly cooled MSRs, where liquid fuel and coolant are in direct contact, are not considered as a promising option anymore, 2) several classification parameters from the extensive list of possible parameters for classifying MSRs³ were not considered at that time, and 3) some recent MSR concepts introduced non-graphite moderating materials,

³ Parameters that have typically been considered for MSR classification are: technological readiness level (low, medium, high), fuel state (liquid, solid), neutron spectrum (thermal, epithermal, fast, hybrid, variable), fuel cooling (in-core, ex-core), contact with coolant (direct, indirect), moderator (graphite, hydrogen based, deuterium based, beryllium based, other), number of actinides containing salts (single-fluid, two-fluids, multi-fluids), salt type (fluorides, chlorides, mixture), salt purpose (fissile, fertile, coolant, moderator), core criticality (critical, subcritical), primary circuit layout (loop, pool), core structure (homogeneous, heterogeneous), use of neutron leakage (reflector, blanket, multi-zone core), neutronics design (burner, converter, breeder, breed-and-burn), make-up fuel (enriched uranium, existing spent fuel, none), fuel cycle (open, partially closed, closed), breeding (Th-U cycle, U-Pu cycle, combination), fuel reprocessing location (in-situ, ex-situ), fuel reprocessing pace (as fast as possible, with years of delay, combination, no reprocessing), fuel refilling / removing (continuously, batch-wise, at once, none), and power level (micro, small, large).

which are different to those used in the past ANP [81] and MSBR [5, 8] research programs at ORNL. Unlike graphite, these moderators are not compatible with the salt and require coating or cladding material to avoid direct contact. Accordingly, they have different technological and safety requirements than those for the MSRs that are moderated by graphite. From this perspective, these concepts are similar with internally indirectly cooled MSRs, where structural material is needed in the core to separate the fuel salt from the dedicated coolant.

4.2. NEUTRONIC CHARACTERISTICS OF MAJOR CONSIDERED NUCLIDES

The taxonomy described in this section divided MSRs into three major classes according to the type of materials present in the core. The fact that graphite is the sole known moderator directly compatible with the salt plays a dominant role in the taxonomy. To provide some insight on the neutronics performance of different MSR concepts, the neutronics characteristics of major nuclides considered for MSRs are discussed here. The selected nuclides are either part of the salt or moderator. Structural materials are not included here. The discussion is sorted by nucleon number and does not follow the actual frequency of appearance of the isotopes in the concepts. The total microscopic cross-section (XS) and the neutron capture probability presented in Figures 1 and 2, respectively, were taken from the ENDF/B-VII.0 XSs library.

FIG. 1. Total microscopic cross-section of selected nuclides.

FIG. 2. Neutron capture probability of selected nuclides.

4.2.1. Hydrogen and Deuterium

Two isotopes of hydrogen, ^1H (referred to herein simply as hydrogen for brevity) and ^2H (deuterium), offer the fastest moderation in nature. Only 14 or 20 collisions are needed to slow down a neutron from 2MeV to 1eV, respectively (see Figure 3). The total microscopic XS of hydrogen is much higher than for deuterium, and both are without resonance (see Figure 1). The higher XS of hydrogen is related to its much higher neutron capture probability (see Figure 2). This characteristic unfortunately deteriorates the neutron economy and disables breeding in thermal MSRs moderated by hydrogen (see Figure 4). According to the same figure, breeding in MSRs moderated by deuterium would be possible. However, any one of these isotopes would require structural material to be separated from the fuel salt. Optionally, they can be embedded in a compound diluted in the salt. In this case, some issues related to chemical stability may arise.

FIG. 3. Number of collisions to slow-down fast neutron from 2MeV to 1eV for selected nuclides.

FIG. 4. Equilibrium k_{inf} as a function of carrier salt type, moderator type, salt share in the core, and salt channel radius.

4.2.2. Lithium 6 and 7

Lithium is considered as a possible cation for both fluoride and chloride salts. It has two isotopes ${}^6\text{Li}$ 7.6% and ${}^7\text{Li}$ 92.4% (percentages are the natural abundances). Since they are light nuclides, only around 50 collisions are needed to slow down a neutron. Compared to hydrogen and deuterium, both Li isotopes have a strong resonance around 260 keV (see Figure 1). There is a huge difference between ${}^6\text{Li}$ and ${}^7\text{Li}$ in the neutron capture probability (see Figure 2). Accordingly, Li cannot be practically applied without isotopic enrichment and removal of ${}^6\text{Li}$. The need is much stronger than in the case of the ${}^{35}\text{Cl}$ and ${}^{37}\text{Cl}$ isotopes, discussed later. The resonance at 260 keV is strongly exposed by fast neutrons, and hence it shifts the neutron spectrum towards lower energy areas and the reactors relying on ${}^7\text{Li}$ have quite low share of fast neutrons in the spectrum. Outside of this resonance, ${}^7\text{Li}$ has very low total microscopic XS and ${}^6\text{Li}$ has very high neutron capture probability. Accordingly, Li is usually not considered as a moderator.

4.2.3. Beryllium 9 and Carbon 12

Compared to Li, beryllium ${}^9\text{Be}$ and carbon ${}^{12}\text{C}$, or actually graphite, are considered as neutron moderators. They both have reasonably high total microscopic XS (see Figure 1). The resonances are small and rather in very high energy range. ${}^9\text{Be}$ and ${}^{12}\text{C}$ need 70 or 90 interactions with a neutron, respectively, to slow it down from 2MeV to 1eV. It is thus important that they have very low neutron capture probability (see Figure 2). Performance of ${}^{12}\text{C}$ is very similar to ${}^9\text{Be}$. However, according to Figure 4, ${}^9\text{Be}$ outperforms ${}^{12}\text{C}$ in moderation capability. Nonetheless, since it is not directly compatible with the salt, it needs cladding or coating, which will deteriorate the neutron economy of the reactor. Graphite is thus considered by more MSR concepts. Its features are thus discussed in a dedicated sub-section, below.

4.2.4. Fluorine 19 and Sodium 23

Fluorine and sodium have only one stable isotope, ${}^{19}\text{F}$ and ${}^{23}\text{Na}$, respectively. ${}^{19}\text{F}$ is considered as the major component of many coolants and fuel salts. It has a similar total microscopic XS as ${}^{23}\text{Na}$, which is considered as a coolant for fast reactors with solid fuel, and as a component of the salt in several MSRs. The XS of these two nuclides have several resonances. In both cases there is one dominating broad resonance, ${}^{23}\text{Na}$ has such resonance with substantial amplitude at 3keV, whereas ${}^{19}\text{F}$ already at around 100keV, but with smaller amplitude (see Figure 1). Similarly, like in the ${}^7\text{Li}$ case, these resonances shift the neutron spectrum to lower energy areas. The occurrence of the ${}^{19}\text{F}$ resonance at higher energies is exposed to higher neutron flux and has thus stronger impact on the spectrum.

Neither ${}^{19}\text{F}$ nor ${}^{23}\text{Na}$ are considered as moderators. For both these nuclides, an interacting neutron needs around 150 collisions to slow down from 2MeV to 1eV (see Figure 3). ${}^{19}\text{F}$ has much lower neutron capture probability (see Figure 2). Accordingly, it is considered also for

thermal MSR with tight neutron economy, such as for thermal breeder reactors. Breeding is nearly impossible in the thermal spectrum with ^{23}Na (see Figure 4).

The frequently applied $^7\text{LiF-BeF}_2$ salt includes a combination of ^{19}F , ^7Li and ^9Be and has some moderating power [8]. This is also the reason that early homogeneous MSFRs were not always called fast [9]. The ^7LiF salt without ^9Be is not moderating per se, but it softens the fast neutron spectrum. The MSFR concept relying on this salt is considered as a fast reactor; however, it has the softest fast spectrum among all fast reactors.

4.2.5. Chlorine 35 and 37

Compared to fluorine, chlorine has two stable isotopes, ^{35}Cl 76% and ^{37}Cl 24%. Both isotopes are heavier than ^{19}F and the interacting neutron thus needs about twice more collisions to slow down from 2MeV to 1eV (see Figure 3). For both chlorine isotopes the XS resonances are quite narrow and the total XS in the fast energy range is lower than for ^{19}F and ^{23}Na (see Figure 1). The neutron capture probability is higher for ^{35}Cl than for ^{37}Cl . However, there are some discrepancies between nuclear data libraries for this nuclide and this probability may be revised. The difference in neutron capture probability between these two isotopes, is nonetheless, much smaller than the difference between ^6Li and ^7Li . It is thus possible to operate a fast chloride breeder with natural chlorine composition.

The combination of low total microscopic XS and high number of collisions needed for neutron slowing down results in very hard spectrum in reactors with these chlorine isotopes. The higher capture probability in the thermal area (see Figure 2) practically disqualifies chlorides for application in the thermal spectrum. Still, they may be considered for cases with relatively relaxed neutron economy, such as in waste burners, which can profit from the chemical features of chlorine.

4.2.6. Graphite moderator

Graphite is the only solid moderator directly compatible with the fluoride salts considered for thermal MSRs. Since it is the most frequently appearing moderator in MSR concepts, it defined one class of the taxonomy, and its features are described here in more detail:

- In power reactors, graphite usually has higher temperature (700°C) than moderators based on ^1H or ^2H . Hence, the Maxwellian spectrum peak is shifted towards higher neutron energies than in reactors using the latter two moderators. This feature, in combination with a fission resonance of ^{233}U slightly above this peak, can result in a positive temperature reactivity effect of the graphite moderator. Nonetheless, it depends on the fuel composition and core size, because the spectral shift is competing with increased neutron leakage, which is caused by the high neutron energy of the Maxwellian spectrum peak.
- Salt volumetric share in the Graphite moderated MSR for optimal neutronics performance is typically around 13%. In the Th-U closed fuel cycle, the reactor often acts as under-moderated. Fuel salt removal from such core results in positive reactivity. This positive reactivity effect is inevitably connected with fuel salt temperature increase and thus with stronger negative reactivity insertion due to the Doppler effect. The total effect is thus negative.

- The ${}^7\text{LiF} - \text{BeF}_2$ salt has some moderating power, but it also has higher neutron absorption than graphite. Therefore, the Graphite moderated MSR could outperform the homogeneous reactor with ${}^7\text{LiF} - \text{BeF}_2$ salt [5][8] from the perspectives of conversion ratio and doubling time.
- To slow down a neutron from 2 MeV to 1 eV with a graphite moderator, 92 collisions are necessary. At the same time, it has 10 times smaller scattering cross-section and 200 times smaller capture cross-section than water. Accordingly, the migration area for neutrons in Graphite moderated MSR can be twice larger than for a Homogeneous fluoride fast MSR. Hence, the treatment of neutron leakage in Graphite moderated MSRs is of high importance. To achieve breeding often blanket, reflector, layout with two zones, bulky core, or a combination of these features must be used.
- The specific neutron moderating process in graphite is sensitive to neutron parasitic captures. This has two consequences. The parasitic capture by the graphite becomes significant before the well-moderated state is reached. Graphite moderated MSRs act always as somewhat under-moderated. The second consequence is that graphite for reactor use should be extremely clean, because impurities, for instance boron, may strongly deteriorate its performance.
- The fissile-to-fertile isotope ratio in an MSR moderated by graphite with the Th-U closed cycle is around 0.015. Together with the optimal 13% salt share in the core, this ratio results in a substantially smaller mass of fissile material load than in a Homogeneous fluoride fast MSR (ratio of 0.15 and 100% salt share in the core). The initial fissile load is thus lower for Graphite moderated MSRs. At the same time, a reactor with very low fissile content in the core is more sensitive to the parasitic absorption of neutrons by fission products.

4.3. DEFINITION OF MAJOR MSR CLASSES AND FAMILIES

There are many parameters and options on how to develop an MSR taxonomy. Similar to animal taxonomy, some of the parameters may or may not be important for its structure. The taxonomy developed herein to classify major MSR concepts is partly inspired by the previous classification [80], but the technological similarities play a more important role in it. The taxonomy covers most of the concepts that have been proposed to date and constitutes from three major classes of MSRs:

- I. Graphite-based MSRs
- II. Homogeneous MSRs
- III. Heterogeneous MSRs

Since some, usually peculiar or smaller power, concepts can be on the border between two classes or not fully congruent with a class definition, a fourth class is introduced to address them:

- IV. Other MSRs

Each class has its own specific features:

- Graphite-based MSR are characterized by graphite presence in the core and employ exclusively fluoride salt and thermal spectrum. Since the graphite is directly compatible with the salt, there is no need for other structural material in the core. This class has the highest TRL level, because of the successful operation of the MSRE, and for some concepts because of their similarity to High Temperature Reactors (HTRs).
- Homogeneous MSR are characterized by the absence of structural materials whose function is to separate the fuel salt from a coolant or a moderator in the core, which is thus filled solely by the fuel salt. Optionally, there can be structural materials (e.g. baffles for flow direction) in the core; however, they do not have the separation function. The TRL of this class depends mainly on the fuel salt, and for some of the MSR using similar salts to those of the MSRE or MSBR, the TRL can be higher than for others.
- Heterogeneous MSR are characterized by the presence of structural material in the core that is separating the fuel salt from a dedicated coolant or from a moderator not directly compatible with the fuel salt. The TRL of this class depends also on the separation material, and can be lower than for the previous two classes because these materials are typically not yet qualified.
- Other MSR, which do not fit the three major classes, have generally the lowest TRL.

4.3.1. I. class

Graphite-based MSR are moderated by graphite and can be further subdivided into two families according to the location of the fuel:

4.3.1.1. 1.1 Fluoride salt cooled reactors

In the family of Fluoride salt cooled reactors, the fuel is in the form of TRISO particles, distributed in the graphite moderator, which most commonly has either plate or pebble shape and the core is cooled by fluoride salt, typically ${}^7\text{LiF-BeF}_2$ eutectic. Concepts from this family are actually classical reactors with solid fuel and share many features with the HTRs; for instance, the very low specific actinides density. The pebble-bed option shares some fuel cycle features of a reactor with liquid fuel that are: 1) continuous refuelling operation, and 2) nearly homogenous core composition obtained as a result of small fuel element stochastic distribution and multiple passes of pebbles in the core. Nevertheless, significant differences exist. As fuel is segregated within pebbles, online isotopic separation is not possible as in liquid fuel but, differently from salt, pebbles can be discerned and discarded according to burnup level. The low specific fuel density and the solid state of the fuel disable sustainable breeding in these reactors. Accordingly, they rely on enriched uranium (typically $>8\% {}^{235}\text{U}$); optionally they can act as TRU burners. Using fluoride salt as coolant enables higher specific power than in an HTR cooled by helium. On the other hand, it results in higher specific decay heat, which cannot be removed only by passive radiation as in the HTR case and need to rely on coolant natural circulation.

4.3.1.2. 1.2 Graphite moderated MSR

The Graphite moderated MSR represent the most developed MSR and use fluoride carrier salt as a diluent for actinide fluorides. Compared to the Fluoride salt cooled reactors, these concepts can reach much higher specific fuel density. Since the fuel is liquid, its continuous treatment is enabled, and breeding can be achieved for selected salt types in the closed Th-U fuel cycle. These reactors can be operated in single-fluid or two-fluids layouts and include

burners fuelled by TRU, converters fuelled by enriched uranium, and breeders in the Th-U cycle.

In both graphite-based families, graphite has limited lifespan due to irradiation. In reactors cooled by fluoride salt, graphite acts as a fuel matrix and is naturally exchanged when the fuel is depleted. The volume of discharged spent fuel is typically much higher than for existing light water reactors, but its form is suitable for final repository. In a Graphite moderated MSR, the graphite has to be regularly replaced solely due to its limited lifespan. It generates voluminous activated waste.

A third family, which is not considered here, can be based on a combination of the previous two options and applicable especially for the transition cycle. The fuel (e.g. enriched uranium) can be embedded in the graphite as TRISO particles, and the fertile material (e.g. thorium) can be diluted in the fluoride salt. However, this design may result in positive coolant-density and coolant-void reactivity effects.

4.3.2. II. class

Homogeneous MSRs are characterized by a fast neutron spectrum and core filled solely by the fuel salt. The two major families are determined by the salt type, fluoride or chloride, as described in the next subsections.

4.3.2.1. II.3. Homogeneous fluoride fast MSRs

The Homogeneous fluoride fast MSRs have probably the second highest TRL level after the Graphite based MSRs. This would be especially valid for the fuel salt used during MSRE or planned for MSBR. However, these particular salts contain beryllium and can be considered as moderating. Therefore, they are usually not applied for Homogeneous fluoride fast MSRs. The TRL level for other fluoride salts may differ between the salt types. These reactors belong to the category of fast spectrum reactors. However, even without beryllium in the salt, their neutron spectrum is one of the softest from fast reactors. They can be operated as breeders in the closed Th-U and U-Pu cycles, as converters fuelled by enriched uranium or as waste burners. For some of the concepts, the solubility of trivalent actinides or lanthanides (e.g. PuF₃) could be the limiting fuel cycle factor.

4.3.2.2. II.4. Homogeneous chloride fast MSRs

The Homogeneous chloride fast MSRs are based on chlorides and historically received less R&D than the MSRs based on fluorides; hence, their TRL level can be lower. These reactors also belong to the group of fast spectrum reactors. However, their neutron spectrum is much harder than for fluoride fast MSRs and it belongs to the hardest spectra between all fast reactors. This is also the most appealing feature of MSRs based on chlorides. It enables efficient breeding in both Th-U and U-Pu closed cycles, and even the operation in an open breed-and-burn U-Pu cycle. The hard spectrum is, however, accompanied by the core transparency for neutrons. Hence, reactors in this family suffer much more from the neutron leakage than the Homogeneous fluoride fast MSRs and are usually larger. At the same time, the solubility of actinides or lanthanides in the fuel salt can be higher than in MSRs that are based on fluoride.

4.3.3. III. class

Heterogeneous MSR are characterized by the presence of structural material in the core that is separating the fuel salt from a dedicated coolant or from a moderator not directly compatible with the fuel salt. The two families in this class are defined by the material that needs to be separated. In one case it is the moderator, which is not directly compatible with the fuel salt, and in the other case it is the dedicated coolant:

4.3.3.1. III.5. Non-graphite moderated MSR

The Non-graphite moderated MSR utilize fuel in the form of fluoride salts diluted in fluoride carrier salt and are characterized by thermal spectrum and presence of structural materials in the core. There are some moderators (e.g., based on beryllium or deuterium) that can assure better neutronics performance than graphite (see Figure 4). Such concepts can be potentially better breeders or waste burners than Graphite moderated MSR. Nonetheless, the performance will strictly depend on the characteristics of the separating structural material. It appears that only silicon carbide (SiC) can preserve the excellent neutronic performance features. However, the lifespan of some of these moderators and structural materials can be significantly larger than graphite, hence producing much less waste, and resulting in the structural material lifespan usually determining the lifespan of the reactor. Since the TRL of this family depends also on the separation material, it can be lower than for other MSR. However, the only operational experience of an MSR besides the MSRE is the ARE which belongs to this class and operated with a beryllium oxide moderator and fluoride fuel salt, separated by Inconel tubes.

4.3.3.2. III.6. Heterogeneous chloride fast MSR

The Heterogeneous chloride fast MSR have their chloride-based fuel separated from the dedicated coolant by the structural material. The coolant can be another salt or liquid metal, typically lead. So far, none of the concepts considers sodium as the coolant. In the salt case, it is usually chloride salt, which does not carry actinides. But in some historical concepts, a blanket salt was used to cool the core. Due to the structural material and dedicated coolant presence in the core, these reactors have higher parasitic neutron absorption in the core than Homogeneous chloride fast MSR. At the same time, the fuel salt does not serve as primary heat transport medium and is therefore not extensively pumped out of the core. Due to the structural materials, which are often not yet qualified, the TRL level of these reactors is lower.

The heterogeneous fast reactors using fluoride fuel salt and dedicated coolant have inferior neutronic performance than both Heterogeneous chloride fast MSR and Homogeneous fluoride fast MSR and are not considered herein.

4.3.4. IV. class

The class "Other MSR" is needed because not all MSR concepts documented in the literature can be covered by the three major classes above. These concepts are typically historical or discontinued, have the lowest TRL level and are rather exotic. Some of the types included in this class are represented by more than one concept. For instance, the Directly cooled MSR. They were proposed to avoid extensive fuel salt pumping through the high-surface heat exchangers or the structural material presence in the active core. The high heat exchange surface is needed, because the salts have low thermal conductivity, which is comparable to water. Since the direct contact leads to strong contamination of the direct coolant, all concepts of this type are discontinued. Another example of more than one concept are the Subcritical

MSRs, which were proposed during the 1990's as a liquid fuel option of the accelerator driven system. Since the only major function of the costly external neutron source is the criticality safety, it can be avoided in inherently safe MSRs. The remaining types are typically based on single rather hypothetical concepts which do not fit one of the six defined families or which are sometimes not meant for large power production. As an example, the gas cooled MSRs or the frozen salt MSRs can be mentioned here.

4.4. MSR TAXONOMY

This Subsection briefly describes the MSR taxonomy. The first two layers of the taxonomy are defined by the three major MSR classes and the six major MSRs families. The three identified major MSR classes are:

- I. Graphite-based MSRs;*
- II. Homogeneous MSRs;*
- III. Heterogeneous MSRs.*

The six major MSRs families are:

- I.1. Fluoride salt cooled reactors (Subsection 4.5);
- I.2. Graphite-moderated MSRs (Subsection 4.6);
- II.3. Homogeneous fluoride fast MSRs (Subsection 4.7);
- II.4. Homogeneous chloride fast MSRs (Subsection 4.8);
- III.5. Non-graphite moderated MSRs (Subsection 4.9);
- III.6. Heterogeneous chloride fast MSRs (Subsection 4.10).

The third level of the taxonomy is the outcome of the detailed family descriptions in Subsections 4.5 through 4.10. Each subsection briefly introduces a family and the family's third layer of the taxonomy. Figure 5 graphically illustrates the three layers of the MSR taxonomy, and the third layer has the heading "Type." The class "Other MSRs" is discussed as a single entity in Subsection 4.11. Appendix V presents a brief description of some current MSR concepts, organized by the six major families.

The six families defined above often share many features across the classes; however, they usually differ within one class. Table 1 offers a comparison of significant parameters for the six families.

FIG. 5. First three layers of MSR taxonomy (tree view).

TABLE 1. COMPARISON OF PARAMETERS FOR THE SIX MAJOR MSR FAMILIES

Class	I. Graphite based MSRs		II. Homogeneous MSRs		III. Heterogeneous MSRs	
Family	I.1 Fluoride salt cooled reactors	I.2 Graphite moderated MSRs	II.3 Homog. fluoride fast MSRs	II.4 Homog. chloride fast MSRs	III.5 Non-graphite moderated MSRs	III.6 Heterog. chloride fast MSRs
Fuel state	Solid	Liquid	Liquid	Liquid	Liquid	Liquid
Spectrum	Thermal	Thermal	Fast	Fast	Thermal	Fast
Salt type	Fluorides	Fluorides	Fluorides	Chlorides	Fluorides	Chlorides
Neutronics performance	Burner, Converter	Burner, Converter, Breeder	Burner, Converter, Breeder	Burner, Converter, Breeder, Breed & Burn	Burner, Converter, Breeder	Burner, Converter, Breeder, Breed & Burn
Actinides	Enriched U, TRU, Th as semi-inert matrix	Enriched U, TRU, closed Th-U cycle	Enriched U, TRU, closed Th-U cycle and U-Pu cycle	Enriched U, TRU, closed Th-U cycle, and U-Pu cycle	Enriched U, TRU, closed Th-U cycle	Enriched U, TRU, closed Th-U cycle, and U-Pu cycle
Irradiation induced issues	Limited burnup of fuel in graphite matrix	Limited graphite moderator lifespan	Limited vessel lifespan	Limited vessel lifespan	Limited vessel and str. material lifespan	Limited vessel and str. material lifespan
Fuel extensive pumping	No	Yes	Yes	Yes	Yes (No if cooled by moderator)	No
Heat transp. medium	Fluoride coolant salt	Fluoride fuel salt	Fluoride fuel salt	Chloride fuel salt	Fluoride fuel salt (or moderator)	Molten salt or lead coolant
Primary heat exchange	In core	Ex. core	Ex. core	Ex. core	Ex. core (In core)	In core

4.5. FLUORIDE SALT COOLED REACTORS

Fluoride salt cooled reactors feature solid fuel, mostly based on coated fuel particles embedded in or surrounded by graphite moderator, cooled by fluoride salt. Similarly, as in the case of the HTR that is gas cooled, the fuel particles can be dispersed in a graphite pebble or concentrated in sealed plates or compacts and surrounded by graphite blocks. The technology of coated fuel particles was already suggested in 1957 [82] for HTR; around the same time when Graphite moderated MSR became a major option for ORNL. Early FHR designs were based on prismatic blocks and TRISO-bearing compacts replicating HTGR prismatic block geometry, but such an option was quickly discarded as the average density of the fuel blocks would be lower than that of the salt causing the blocks to float and making some operations, such as refuelling, very challenging. Designs featuring plate, pebble or stringer fuel have instead gained traction.

Fluoride salt cooled reactors combine two previously existing technologies: reactor coolants based on molten fluoride salts operating at high temperature and low pressure developed in the USA aircraft nuclear propulsion program of the 1950s, and the subsequent breeder reactor program that was based on molten salt, and solid fuel consisting of coated particles (tristructural isotropic or TRISO) embedded in a graphite matrix originally developed for reactors operating at high temperature and cooled by gas in Member States such as Germany and USA. As fuel is based on TRISO particles embedded in or surrounded by a graphite moderator, Fluoride salt cooled reactors are characterized by a soft neutron spectrum. This feature restricts the choice for coolant to neutronically favourable fluoride salts. Among those, the ${}^7\text{LiF}-\text{BeF}_2$ (2:1) eutectic remains the preferred option based on neutronics (if enriched in ${}^7\text{Li}$ at 99.995% or higher percentage) and thermophysical properties. Although graphite is the main moderator, the salt's light components also contribute to neutrons slowing down. Moreover, to obtain negative coolant temperature or coolant void reactivity feedback, it is, indeed, necessary that salt moderation is not negligible. Therefore, the coolant scattering cross-section should have a stronger impact on the neutronics than the capture cross-section. When the salt temperature increases (causing the salt's density to decrease) or when a void appears in the salt, two competing reactivity changes occur: a positive change due to reduced absorption of neutrons in the salt, and a negative effect due to the spectrum hardening from reduced moderation in the salt. For the latter effect to be dominant and the overall feedback to be negative, neutron moderation in the salt must be sizable and neutron capture minimal [83]. Therefore, 1) only fluoride salts are suitable for the coolant, 2) ${}^6\text{Li}$ is excluded for its high capture cross-section, and 3) ${}^9\text{Be}$ is included for its high scattering cross-section.

As already mentioned, the low specific fuel density and the solid state of the fuel disable sustainable breeding in these reactors. Accordingly, they rely on enriched uranium (typically $> 8\%$ ${}^{235}\text{U}$); optionally they can act as TRU burners. Using fluoride salt as coolant enables higher specific power than in an HTR cooled by helium. On the other hand, it results in higher decay heat that is removed relying on coolant natural circulation rather than passive radiation as in the HTR case.

4.5.1. Major reactor types

The major reactor types in this family are described in the following text, and major concepts introduced in Table 2.

4.5.1.1. *Salt cooled reactor with pebble bed fuel*

A pebble bed reactor concept cooled by salt is first mentioned in a paper from 1983 [66] although no development followed. Currently, the most common pebble bed type variant is the Fluoride-cooled High-temperature Reactor (FHR) initially proposed in the early 2000s [84].

The design based on a pebble bed was developed at the University of California, Berkeley (UCB) as part of a multi-university project (MIT, UCB, and University of Wisconsin-Madison). The UCB concept, named Mk1 PB-FHR [85], uses annular pebbles that are 3 cm in diameter and contain fuel (TRISO particles) in a thin layer at the outer region of the pebble to limit the operating temperature of the particles. Pebbles are slightly buoyant in the salt, and therefore they are inserted from the bottom of the core and extracted at the top, which is opposite to the direction in pebble beds cooled by gas. Typical of a system based on a pebble bed, the Mk1 PB-FHR features online refuelling that enables low reactivity excess. Commercial deployment of a PB-FHR is being pursued by Kairos Power LLC and its proprietary design is known as KP-FHR.

Due to the low heavy metal content of fuels based on particles, FHRs require 10% to 20% uranium enrichment. The better heat transfer capabilities of the salt allow FHRs to achieve a power density that is double that of reactors cooled by gas (about 20 W/cm³).

4.5.1.2. *Salt cooled reactor with fixed fuel*

Multiple molten salt reactor designs have been proposed that rely on stationary fuel. ORNL has pursued the use of plate fuel embedding TRISO particles in the development of the Small Modular Advanced High Temperature Reactor (Sm-AHTR) cooled by a fluoride salt [86]. This was mostly a feasibility study and no further development has occurred in recent years. An early study evaluated using conventional fuel pellets or stringer fuel and found such option viable although more in depth studies were not pursued [87]. A recent design variant based on the British Advanced Gas-Cooled Reactor (AGR) employs cylindrical fuel stringers made of graphite within which are channels for coolant and channels containing fuel alternate [88]. The fuel stringers contain TRISO bearing compacts and are housed in a graphite core. The AGR-FHR aims to benefit from the long operational experience of AGR and to adapt its technology of refuelling at high temperature.

TABLE 2. FAMILY I.1. - FLOURIDE SALT COOLED REACTORS

Class	Family	Type	Concept name	Designer	Appear in year	Ref.
I.	1.	Salt cooled reactor with pebble bed fuel	FCSR	Kurchatov Institute	1983	[66]
			PB-FHR, KP-FHR	UCB, Kairos Power	2016	[85]
		Salt cooled reactor with fixed fuel	AHTR, SmAHTR	ORNL	2010	[86]
			AGR-FHR	Forsberg	2019	[88]

4.6. GRAPHITE MODERATED MSRS

The reactor concepts belonging to this family use graphite as moderator, hence the neutron spectrum is thermal. The fuel is liquid in the form of actinide fluorides dissolved in a fluoride carrier salt or their mixture, for instance, BeF_2 , ${}^7\text{LiF}$, or NaF . The fuel has simultaneously the coolant function, so it not only conveys the fission heat out of the core, but also cools the graphite moderator. From the perspective of the fuel cycle, this reactor can be operated as a breeder in a closed Th-U fuel cycle, as a converter fuelled by enriched uranium, or as transuranics (TRU) burner in open or closed fuel cycle. This reactor type has a prominent position from a historical perspective, and the highest experimental and theoretical effort so far was invested for this technology.

The fuel and the moderator are in direct contact. The capability of graphite to withstand this contact was recognized in the early pioneering time of the nuclear era. During the Aircraft Nuclear Propulsion (ANP) program between 1947-1961 (e.g. [89]) at ORNL, the reactors moderated by graphite were excluded as an option for the nuclear-powered jet engines because of their size. The associated shielding weight penalty was too big [81]. Nonetheless, the benefits of graphite features were recognized, and graphite was used or proposed to be used: as outer shielding of the reactor moderated by Be [90, 91], as graphite packed seals [92], as graphite powder for extinguishing material [93], as vessels lined with graphite for fuel salt preparation [93], as graphite anode for electrochemistry in chlorides [94], or in the compound $\text{UF}_6\text{-C}_7\text{F}_{16}$ [95]. Tests were done within the ANP program for: compatibility of graphite of high density with other materials [96], and changes in a chemical system's behaviour with graphite presence [97]. Many other studies were done later (e.g. in Ref. [98]), including carburization of Inconel and INOR-8, irradiation time and radiation damage of graphite, and graphite permeability for molten salts.

The three major requirements for a graphite moderator, which are still valid today, are recapitulated in Ref. [99]:

- it must withstand neutron irradiation for enough time;
- it must have pores small enough that capillary forces exclude fuel salt; and
- it must have a low enough permeability to gases to keep down the absorption of xenon.

ORNL decided to develop a civilian MSR breeder around 1957. In Refs [8, 5], the Th-U cycle is recognized as an alternative to the U-Pu cycle, and a fluoride salt mixture consisting of BeF_2 ,

${}^7\text{LiF}$ (or NaF), UF_4 and ThF_4 was proposed for both a Homogeneous fluoride fast MSR and a Graphite moderated MSR. The performance of both reactor concepts was comparable; the moderated reactor based on a ${}^7\text{LiF}$ - BeF_2 salt even slightly outperformed the fast reactor [100, 101] from the perspectives of conversion ratio and doubling time. At that time, other moderators were excluded because they are not compatible with the fuel salt and the parasitic neutron absorption by their respective claddings would disable sustainable breeding in the Th-U cycle [8, 100]. The Th-U cycle generates the intermediate isotope ${}^{233}\text{Pa}$, which decays with 27 days half-life to ${}^{233}\text{U}$ and causes parasitic capture of neutrons. It is especially pronounced in the thermal spectrum. Hence, it is desirable to remove this isotope or to minimize this capture in a graphite moderated MSR. To achieve breeding in the Th-U closed cycle with a graphite moderated MSR, several design solutions are needed: leakage minimization with blanket, reflector, under-moderated outer core zone, bulky core, or a combination of these features; intensive salt purification from fission products; and minimization of ${}^{233}\text{Pa}$ parasitic capture of neutrons.

Other considerations related to graphite that are relevant for MSR application are the following:

- Potentially positive effect on reactivity due to temperature;
- Large thermal inertia;
- Large cores;
- Slightly under-moderated cores;
- Limited lifespan of graphite due to irradiation;
- Penetration by salt and gases;
- Shrinking and growing under irradiation;
- Radioactive graphite waste;
- Reactor can operate with a low fissile inventory;
- Need for intensive salt cleaning and ${}^{233}\text{Pa}$ removal in breeder reactor concepts;
- Complexity of reactor concepts using two fluids;
- Graphite reflector can reduce neutron fluence of the reactor vessel;
- Graphite is neutronicly compatible with moderating ${}^7\text{LiF}$ - BeF_2 salt;
- Presence of BeF_2 decreases salt melting temperature.

4.6.1. Major reactor types

The major reactor types in this family are described in the following text, and major concepts introduced in Table 3.

4.6.1.1. *Single-fluid Th-U breeder*

Since it is desirable to remove ${}^{233}\text{Pa}$ and fission products because of their parasitic neutron capture, a design with a single fluid was proposed when new techniques were developed in 1971 for extraction of ${}^{233}\text{Pa}$ using liquid bismuth and of rare earth fission products from a salt with mixed fertile and fissile isotopes [102]. The resulting MSBR design (see illustration in Figure 6) with a single fluid [103] had two zones with salt to graphite ratio of 13% and 37% in the central fissile and in the peripheral fertile zones, respectively. This design became an inspiration for an entire family of MSBR concepts, such as FUJI [104, 105], AMSTER (Actinides Molten Salt TransmutER) [106], and TMSR (Thorium Molten Salt Reactor) [107]. The complexity of designs with two fluids can be avoided by using a single fluid. The major drawbacks, typical for an MSR moderated by graphite, are not avoided: the limited lifespan of graphite, requirement of ${}^{233}\text{Pa}$ extraction, and the need for intensive removal of fission

products. Since the salt share in the peripheral fertile zones was around 37% in the ORNL design, the graphite pebbles were considered as an alternative in one of the concepts [108]. Nonetheless, pebbles are not suitable for the core region, which requires much lower salt share to obtain optimal fuel-to-moderator ratio.

4.6.1.2. *Two fluids Th-U breeder*

Historically, the first MSR concepts using graphite were based on a layout with two fluids. The separation of the fertile salt containing Th from the fissile salt has some neutronic and reprocessing advantages. Nonetheless, for breeding to occur this separation can only be implemented with material with low absorption of neutrons, for instance graphite. The Molten Salt Breeder Reactor (MSBR) of 2,225 MW(th) was a reference Th-U breeder concept that was initially proposed with two fluids until 1968 at ORNL when design efforts transitioned to a single fluid design [109, 110, 111]. The fuel salt was separated from the blanket salt by graphite tubes. A similar solution is nowadays adopted by the Liquid-Fluoride Thorium Reactor (LFTR) concept [112]. The presence of two fluids in the core results in complexity of the design. This complexity was underlined when the limited graphite lifespan due to irradiation was recognized. Accordingly, regular and remote graphite replacement is required to assure the tightness of the barrier between the two fluids [113]. The SSR-Th concept in Ref. [114] is also included within this reactor type, even though it uses structural material to separate the fissile and fertile salts.

4.6.1.3. *Uranium converters and other concepts*

The MSRE was designed as a burner of highly fissile fuel (e.g. [19]) with a very high ratio of fissile actinides. Even though it was developed in the frame of the MSBR program, thorium was never used in this reactor. Ref. [113] proposed the Molten Salt Breeder Experiment (MSBE) as a reactor concept with thorium and two fluids. Later, a Molten Salt Demonstration Reactor (MSDR) with 300MW(e) was discussed with an industrial consortium and drafted in [115]. The last major concept from this category proposed by ORNL was the Denatured Molten Salt Reactor (DMSR). It addressed the proliferation risk of ^{233}U by ^{238}U addition. The ^{233}Pa parasitic neutron capture was minimized by low specific power, without the need for removing this isotope. The low specific power also enabled 30 years of operation without exchanging the graphite [31]. As a trade-off, the active core diameter was 830 and 1000 cm without and with reflector, respectively. The enormous salt volume in the large DMSR core provided fuel reserve for 30 years of operation without extensive removal of fission products.

Nowadays there are several converter concepts moderated by graphite, such as FUJI [52], Integral MSR (IMSR) [116], and ThorCon [117]. Giving up on the self-sustaining breeding in a closed cycle provides a design margin, which can be used for optimizing other parameters. Accordingly, the IMSR burns enriched uranium in a compact integral layout. The lifespan of graphite, around 7 years, matches the lifespan of the vessel material. ThorCon's description includes both enriched uranium and thorium. Hence the same denaturation as in DMSR can be used; however, the core would probably be more compact. The FUJI MSR designed in Japan had several versions, and one was dedicated to a high burnup cycle based on uranium. The SSR-U reactor in Ref. [114] is also included within this reactor type, even though it uses structural material to separate the fuel and coolant salts.

TABLE 3. FAMILY I.2. - GRAPHITE MODERATED MSRS

Class	Family	Type	Concept name	Designer	Appear in year	Ref.
I.	2.	Single-fluid Th-U breeder	MSBR	ORNL	1971	[103]
			FUJI	Furukawa	1992	[104] [105]
			AMSTER	Électricité de France (EdF)	2000	[106]
			TMSR	SINAP	2013	[107]
		Two-fluids Th-U breeder	MSBR2f	ORNL	1970	[109] [110] [111]
			LFTR	Flibe Energy	2011	[112]
			SSR-Th*	Moltex	2018	[114]
		Uranium converters and other concepts	MSRE	ORNL	1965	[19]
			MSBE	ORNL	1967	[113]
			MSDR	ORNL	1971	[115]
			DMSR	ORNL	1980	[31]
			FUJI	Furukawa	1987	[52]
			IMSR	Terrestrial Energy	2013	[116]
			ThorCon	ThorCon	2015	[117]
		SSR-U	Moltex	2018	[114]	

* The SSR-Th and SSR-U as reactors moderated by graphite belong to the respective reactor types. However, they also use metallic structural materials to separate the fissile and fertile salts or the fuel and coolant salts, respectively. As such, they have some common features with the heterogeneous MSRs.

FIG. 6. Simplified figure of graphite moderated two-zone MSBR design proposed at ORNL in 1971.

4.7. HOMOGENEOUS FLUORIDE FAST MSRS

The reactor concepts belonging to this family do not use any moderator making the neutron spectrum fast. The fuel is liquid in the form of actinide fluoride and the carrier salt is usually ${}^7\text{LiF}$ even if other options with BeF_2 , NaF , and KF are possible. The fuel mixture acts also as coolant, and thus the homogenous fluoride fast MSR belongs to the externally indirectly cooled MSRs. The main options for the fuel cycle are based on Thorium (Th-U and Th-TRU) allowing the reactors in this category suitable for breeder and burner applications. Utilization of enriched uranium or of the U-Pu cycle is also possible, but may face the issue with trivalent fluorides, for example, PuF_3 solubility limit.

Homogenous fluoride fast MSRs have one of the softest fast spectra and the U-Pu cycle especially suffers from this fact. The Th-U cycle is less dependent on the spectrum and acts thus as a reference fuel cycle for these reactors. The combination of liquid fuel and fast

spectrum allow for both waste incineration of the actinides that are long lived and breeding for resource sustainability. This possibility brings different benefits in the reactor design, safety and operation. Firstly, the absence of graphite or other any moderator material leads to a simplification of the layout of the fuel circuit, typically characterized by the absence of structural internals, providing advantages from the neutronics and thermohydraulic point of views. However, this absence also reduces thermal inertia of the core and results in a higher actinides load. Secondly, due to the reduced capture cross-section of the fission products (FPs) and due to the increased fissile fuel share in the fast spectrum, these reactors are less sensitive to FPs and there is no need for frequent salt treatment.

From a safety point of view, when compared to thermal reactors, these fast MSR are characterized by i) highly negative reactivity coefficient, both for the salt void (thermal expansion) and the temperature Doppler effect; ii) a small reactivity swing, iii) a lower tritium production iv) a much shorter lifetime of prompt neutrons, and v) a lower thermal inertia. On the other hand, a fast reactor has a hard neutron spectrum because many of the neutrons have high energies, and this spectrum causes higher irradiation damage in terms of displacements per atom (dpa) to those structural materials that are in direct contact to the salt and not protected and shielded by a graphite reflector.

The fuel salt carrier employed by this MSR family is characterized by mixtures based on fluoride (typical cations are alkali or alkaline earth ions) due to their favourable features in terms of melting and boiling points (compatible with reactor design and structural material limitations), low capture cross-section, and high solubility of actinide and lanthanide fluorides. The latter property is particularly important in this family since the fast spectrum requires a much larger fissile and fertile inventory than thermal MSRs. Fluoride salts have good thermophysical properties with respect to other candidates (general candidates or salt candidates) in terms of density, heat capacity, and thermal conductivity. However, they are not as good as liquid metals, which are an excellent medium for heat exchange, especially in passive decay heat removal (DHR). With respect to other candidate salts for a fast MSR (mainly chloride salts), fluoride salts can rely on a relatively extensive experience given their use in the molten salt reactors moderated by graphite operated in the past as the MSRE and in other experimental tests (e.g., SALIENT irradiation tests at NRG, or thermohydraulic tests as Liquid Salt Test Loop at ORNL, FFFER and SWATH at CNRS Grenoble, ADETTE technological loop programme in Czech Republic, and CIET facility at UC Berkeley), even if their behaviour in an environment with fast neutrons is not well established. In addition (always with respect to the chloride salt option), reactor concepts based on fluoride salts show less hard spectrum, a proven compatibility with structural materials (e.g., Hastelloy-N), less chemical issues (e.g., Chlorine isotope separation is required to decrease the production of ^{36}Cl but it is not ready at industrial scale), and no sulphur production. On the other hand, using a fluoride salt bearing lithium or beryllium can be a concern due to tritium generation, and Li also needs isotope separation to decrease tritium production and increase the neutron economy.

Different options for fuel cycles are available in this family, according to the reactor operation mode, that is, whether the reactor works as TRU burner, converter, or breeder. Since the solubility limit of trivalent fluorides often defines the applicable carrier salt, Homogenous fluoride fast MSRs can be generally divided into Th-U breeders and reactors containing Pu. In the second case, it can be designated as a TRU burner or breeder in closed U-Pu cycle. At the same time, ^{233}U is not available in sufficient amount and Th-U breeders will need to be started

with enriched uranium, reactor grade plutonium, or a mixture of the two. The reactor design for such fuel cycle transition will be classified according to the applied fuel type.

4.7.1. Major reactor types

The major reactor types in this family are described in the following text, and major concepts introduced in Table 4.

4.7.1.1. Fluoride fast Th-U breeder

As mentioned in Section 2 on the history of MSR technology, the fast breeder reactor based on fluoride salts and the Th-U cycle was proposed by a group of students at Oak Ridge School Of Reactor Technology (ORSORT) in 1953 [9]. It is a predecessor of the Molten Salt Fast Reactor (MSFR) design. However, it was based on LiF-BeF₂ carrier salt and reflected by graphite. Accordingly, the average neutron spectrum was epithermal to fast and the performance, especially the doubling time, was worse than for Graphite moderated MSRs. Two-fluid layout of Homogeneous fluoride fast MSRs concept is presented in ORNL report from 1958 [118]. It is a very representative concept, showing the typical core dimensions, e.g. 2 m diameter, and fuel-blanket layout used by many later concepts (see Figure 7). However, this concept was published around the time when Graphite moderated MSR was selected as a reference concept at ORNL and thus abandoned.

Most advanced concept of this type is the Molten Salt Fast Reactor (MSFR), developed in the framework of several European Projects (EVOL, SAMOFAR, SAMOSAFER). It is a 3,000 MW(th) reactor, employing LiF as the salt carrier with ThF₄-²³³UF₄ or ThF₄-UF₄-(TRU)F₃ as fuel options. However, there is also the Indian Molten Salt Breeder Reactor (IMSBR), developed at Bhabha Atomic Research Centre (India). This name identifies reactor concepts having a loop or a pool. They are 850 MW(e) reactors, employing LiF as the salt carrier and ThF₄-²³³UF₄ as fuel.

4.7.1.2. Pu containing fluoride fast reactor

Since the solubility of PuF₃ is limited in LiF-ThF₄ and LiF-UF₄ salts, usually other carrier salts are envisioned for Pu containing reactors. The same can be valid for TRU burners which are included in this type. From the possible cations ⁷Li, Na, and K, lithium has the lowest capture cross-section. At the same time, ⁷Li has significant resonance of the scattering cross-section at 260 keV and it is the lightest element from the comparison. Therefore, it is softening the neutron spectrum. It can be the best option for the closed Th-U cycle, which is more sensitive to parasitic capture than to spectrum softening. In the U-Pu cycle case it is the opposite. Accordingly, when lithium is avoided and the reactor spectrum becomes harder, the neutron balance of the U-Pu cycle improves. However, the increased parasitic neutron absorption of the salt slightly deteriorates this improvement.

The closed U-Pu cycle was utilized by Fast Molten Salt Reactor (FMSR) developed at VNIINM (Russia). It is a 3,200 MW(t) reactor concept, employing FLiNaK as salt carrier and UF₄ – PuF₃ as fuel [119]. Similar concept with closed U-Pu cycle and FNaK carrier salt was analysed in [120]. The major concept of a Pu containing fluoride fast reactor is the Molten Salt Actinide Recycler and Transformer (MOSART) [121], developed at Kurchatov Institute (Russia). It is a 2,400 MW(th) reactor, employing ⁷LiF-BeF₂ or NaF-⁷LiF-BeF₂ as salt carrier and (TRU)F₃ as fuel.

TABLE 4. FAMILY II.3. - HOMOGENEOUS FLUORIDE FAST MSRS

Class	Family	Type	Concept name	Designer	Appear in year	Ref.
II.	3.	Fluoride fast Th-U breeder	Fused Salt (Fast) Breeder Reactor (FSBR)	ORSORT	1952	[9]
			Two-region, homogeneous MSR	ORNL	1958	[118]
			MOSART (breeding option)	Kurchatov Institute	2001	[122]
			MSFR	CNRS	2005	[59]
			IMSBR	BARC	2015	[123]
		Fluoride fast Pu-fuelled reactor	MOSART (burner option)	Kurchatov Institute	2001	[121]
			FMSR	VNIINM	2015	[119]
			Molten Salt Fast Breeder Reactor (MSFBR)	Hirose	2016	[120]

FIG. 7 Representative figure of Homogeneous fluoride fast MSRs concept proposed by ORNL in 1958 [118].

4.8. HOMOGENEOUS CHLORIDE FAST MSRS

The Homogeneous chloride fast MSR are very similar to Homogeneous fluoride fast MSRs. The obvious major difference is the use of chloride salts. Different options in terms of carrier salts are available [32], ranging from the alkali (Na, K) and alkaline earth (Ca, Mg) compounds to CuCl and ZnCl₂. This family is characterized by a hard fast spectrum given the absence of light nuclei in the carrier salt, such as Li and Be, usually employed in concepts based on fluoride. This feature is exploited in the fuel cycle where the most selected option is the breeder U-Pu cycle, or less frequently U-Th cycle. Burner and converter applications are also possible, as well as the self-sustaining breed-and-burn U-Pu cycle, which avoids the need of fuel enrichment, reprocessing and separation of actinides.

The use of chloride salt was one of the first options considered during the first phase of the development of MSR concepts. Being alternative to the salts based on fluoride, this option was discarded mainly in favour of fluoride, in addition to having the problem related to the activation of ³⁵Cl into ³⁶Cl, an isotope that is long lived and a 0.7 MeV beta emitter. A solution to this radiotoxicity issue is the chlorine isotopic separation with enrichment in ³⁷Cl, but this technology was not considered mature at that time. As result, R&D efforts were concentrated in fluoride carriers, making experimental data on chloride very limited. In the 1950s, different concepts were proposed based on the use of chloride in fast MSRs considering also the homogenous option and a U-Pu cycle [7, 124]. Later chlorides salts were considered also for pyro-reprocessing [41] and studied in several countries [32, 35, 125, 126] during the 1960s.

In homogenous chloride fast reactors, the active core is composed of fuel salt with no structural material that may affect the neutronics feature. Being the power directly released in the fuel salt, the heat exchange with a secondary salt takes place outside the active region leading to a loss in delayed neutron precursor and an overall increase of the required fissile inventory due to the salt volume outside the core active region. For this purpose, the higher solubility for the actinides granted by the chloride salts allows for an increased actinides load capacity with respect to the fluorides case. From a neutronics point of view, the chloride salts are characterized by the presence of two stable isotopes, ^{35}Cl (75.76%) and ^{37}Cl (24.24%). In addition to the production of ^{36}Cl , the ^{35}Cl has a higher capture cross section than ^{37}Cl , leading to the need of isotopic separation in a chloride fast MSR. The low scattering properties of ^{37}Cl , in combination with the low neutron absorption, provide a hard spectrum to this kind of MSR. On the other hand, this results also in a large neutron mean free path, requiring a reflector or a blanket for counterbalancing the neutron transparency. As for the issues related to chemistry, such as corrosion, compatibility with structural material, and stability of chloride salts, they require further investigation. From a thermohydraulic point of view, the chloride salts share similar characteristics with fluoride salts, having good heat transfer properties and showing small differences in terms of heat capacity, density and viscosity. In terms of melting point, the chloride salts show a slight lower value than fluoride salts.

4.8.1. Major reactor types

The major reactor types in this family are described in the following text, and major concepts introduced in Table 5.

4.8.1.1. Chloride fast breeder reactor

Reactors of this type rely on fuel recycling in closed U-Pu fuel cycle. The Th-U fuel cycle would be also possible. However, the chlorides transparency for neutrons requires a high neutron multiplication factor to keep the core size reasonable. Since the multiplication factor of the Th-U cycle is lower, the operation is possible, but the core size would be much larger.

The possibility to close the U-Pu cycle in Homogeneous chloride fast MSRs was assessed as early as in 1952 [124]. It was followed by several studies (refer to Table 5). All these concepts were relying on ^{37}Cl enrichment to improve the neutron economy. However, there were also some designed for natural chlorine [37]. Similarly, there is one concept designed for the combined Th-U and U-Pu breeding cycle [36]. Another reactor concept dedicated to waste burning [127] is included here, even though it is not a breeder. Nonetheless, it is proposed to be coupled to a U-Pu cycle breeder. The most recent concept was proposed in 2004 by EDF and is labelled as REBUS [62, 63]. All the concepts mentioned here rely on NaCl carrier salt.

4.8.1.2. Chloride fast B&B reactor

The Homogeneous chloride fast MSRs have also the capability to operate in open breed-and-burn (B&B) fuel cycle. In this specific cycle, fuel is not recycled. The reactor is operated on natural or depleted uranium, because the excess of fissile fuel bred in the core is equal or higher than the amount of fissile fuel discharged from the core. Hence, fuel recycling is not needed. The principle of breed-and-burn cycle is known for solid fuel reactors since decades ago. Nonetheless, first publications related to breed-and-burn cycle in MSR are from 2015 [128, 129]. Since the breed-and-burn mode requires much higher neutron economy than the breeder

in closed fuel cycle, the size of the breed-and-burn reactor should be much larger for comparable blanket or reflector conditions. The neutrons saved by the decreased leakage are utilized for breeding of the fuel amount, which is finally discharged with the fuel. The list of breed-and-burn is in the Table 5.

4.9. NON-GRAPHITE MODERATED MSRS

The reactor concepts belonging to this family are a cohort of moderated MSR designs, where the neutron spectrum is thermal, and characterized by using non-graphite moderators. From the perspective of the fuel cycle, these reactors can be operated as a breeder in a closed Th-U fuel cycle, as a converter fuelled by enriched uranium, or as a TRU burner in an open or closed fuel cycle. These designs separate the fuel salt and moderator due to chemical incompatibility, and in some cases thermally insulate the moderator from the fuel salt. This large MSR family used diverse moderator types, but many of these MSR designs were not actively developed after the termination of the ANP program (1947-1961) at ORNL.

Broadly, this family can be split up into three main types:

- Solid moderator heterogeneous MSRs. The moderator is in the form of metals, metal oxides, metal hydrides, or metal carbides. This group excludes MSRs moderated by graphite, which is considered a separate family, as described in Subsection 3.7;
- Liquid moderator heterogeneous MSRs. The moderator is in the form of circulating or stagnant liquid, for example, light water, heavy water, metal hydroxides, or metal deuterioxides. This group includes reactors cooled by circulating the fuel salt or the moderator.
- Hybrid moderator heterogeneous MSRs, both employing solid and liquid moderators. Since this type is discontinued, it is included in Other MSRs class.

Homogeneous mixtures of fuel salt and carrier salt, referred to simply as 'fuel salt', is not considered moderating enough to be part of this family but is, instead, considered a fast reactor; see Subsections 4.7 and 4.8 on Homogeneous fluoride and chloride fast MSRs, respectively. However, fluoride fuel salts, especially those containing beryllium difluoride, might be characterized as epithermal rather than fast, and contribute significant self-moderation in otherwise unmoderated designs.

TABLE 5. FAMILY II.4. - HOMOGENEOUS CHLORIDE FAST MSRS

Class	Family	Type	Concept name	Designer	Appear in year	Ref.
II.	4.	Chloride fast breeder reactor	Fast Converter	Goodman et al.	1952	[124]
			Fused Salt Fast Breeder	Bulmer et al.	1956	[7]
			Homogeneous chloride-fueled fast reactor	Nelson et al.	1967	[130]
			Molten chloride Salt Fast Reactor	Smith et al.	1974	[42]
			Thorium-Uranium Fast/Thermal Breeder	Taube	1974	[36]
			High-Flux Fast Molten Salt Reactor	Taube	1975	[127]
			SOFT	ETR	1980	[37]
			Molten Chloride Fast Reactor (MCFR) Th-U - Molten Chloride Fast Reactor	Ottewite	1982	[38]
			REBUS	EDF	2004	[62] [63]
		Chloride fast breed-and-burn reactor	B&B MCFR	Hombourger	2015	[128] [131] [132]
			MCFR	TerraPower	2015	[129]
			Molten Chloride Salt Fast Reactor (MCSFR)	Elysium Industries	2017	[133]
			B&B MCFR in multizone	Raffuzzi and Krepel	2020	[134]
			B&B MCFR with baffles for flow direction	De Oliveira and Hombourger	2020	[135]

Concepts with a homogeneous mixture of solid fuel and moderator in the form of a molten salt, such as uranium oxide particles suspended in molten hydroxide, that is, concepts based on slurry, and concepts with heterogeneous solid fuel and same type of moderator, such as solid fuel assemblies or pebble beds cooled and moderated with, for example, molten hydroxide, are characterized as reactors cooled by molten salt, and thus not part of this family.

Since the ANP program was aiming at jet engines powered by nuclear energy, reactors with high outlet temperature, compact dimensions, reasonable weight and sufficient control stability

were required. The compact dimension requirement conflicts with the reactor breeding capability for reactors without blankets and the fertile isotope ^{232}Th was excluded and ^{238}U minimized.

The different moderator types are listed also in Ref. [136], including the direct mixture of liquid moderator with a compound bearing fuel. They are evaluated from a breeder perspective and have either low solubility for actinides or include nuclides that absorb neutrons. These characteristics deteriorate the breeding performance in a thermal spectrum.

When the ANP program was terminated, MSR research at ORNL shifted towards developing reactors that produce power, aiming at thermal breeder reactors. At that time, other moderators were excluded because they were not compatible with the fuel salt and the parasitic neutron absorption of the respective cladding would disable sustainable breeding in the Th-U cycle [8, 100].

Recently, the performance of selected moderators and fuel salts was evaluated in the closed Th-U fuel cycle [131]. When the cladding material is neglected, moderators based on beryllium and deuterium, namely beryllium metal, beryllium oxide ceramic, and heavy water, neutronicly outperform the graphite (see Figure 8). The study did not include $^7\text{LiOD}$, which is likely to have similar neutronic performance to heavy water. Moderators based on hydrogen, even with the neglected cladding, are not applicable for sustainable breeding. When the metallic cladding containing Hastelloy or SS316 is considered, moderators based on beryllium and deuterium are not applicable for sustainable breeding. The only applicable cladding material that allows sustainable breeding is SiC. However, both beryllium and heavy water with SiC cladding neutronicly outperforms graphite moderators breeding performance, making them the highest neutronicly performing thermal spectrum molten salt reactors. For a power reactor beryllium will likely require internal cooling while heavy water can be circulated for cooling.

The now defunct Transatomic Power considered a solid moderator made of zirconium hydrate (ZrH) in direct contact with the fuel salt and possibly chemically stabilized with lithium hydride (LiH) and yttrium hydride (YH) [137].

The limited life of graphite under radiation, issues related to ingress of fuel salt and gaseous species, and possibly improved neutron economy and lower fissile inventory has led to liquid moderator being the focus of recent MSR developments. Liquid moderators principally have infinite lifetime, similar to the fuel salts, resulting in the cladding material lifetime setting the overall possible reactor lifetime, e.g. SiC cladding has significantly larger resistance to reactor irradiation than fuel salt compatible metal alloys.

4.9.1. Major reactor types

The major reactor types in this family are described in the following text, and major concepts introduced in Table 6.

4.9.1.1. Solid moderator heterogeneous MSRs

The ANP program studied and considered many non-graphite solid moderators together with molten salt fuels, including beryllium (Be) and beryllium oxide (BeO) [81], magnesium oxide

[138], several hydrides and hydrates [139], and beryllium carbide [140]. Thermal stress considerations limited BeO moderator application for reactors with power density below 0.5 kW/cm^3 [81]. Metal hydrides were investigated and found to be stable under irradiation, but there was none with satisfactory physical properties during the early ANP program [81], and an unpublished zirconium hydride moderated MSR concept with the fuel salt separated from direct contact with the moderator via tubes was suggested by Pratt & Whitney [138]. The compatibility of graphite and molten salt was identified early on, but a reactor that was reflected and moderated by graphite required bigger cores than other concepts, and the penalty caused by the shielding weight made them especially undesirable for nuclear powered aircraft [81]. However, several studies of civil reactors that were purely moderated by graphite were done as early as 1953 [138], and these reactors were being substantially researched by the late 1950s [140].

In the frame of the ANP project, the Aircraft Reactor Experiment (ARE) was operated successfully for 9 days in 1954. It was a small reactor with 2.5 MWt power. The core outlet temperature at steady state was ranging up to about 860°C . The fuel was in the form of a mixture of molten salt fluorides (NaF , ZrF_4 , and UF_4) and the reactor was moderated by BeO. The thermal stresses in the moderator were limiting the reactor power [81].

One of the major concepts proposed by ANP was a compact MSR core, also referred to as 'Fireball' [138], with Be reflector and Be central block, both cooled by a liquid NaK alloy. It was proposed as the Circulation Fuel Reactor Experiment (CFRE) and later as Aircraft Reactor Test (ART) [95], which was not constructed. The moderator was separated from the fuel salt by cladding in ARE and ART.

The concept of the now defunct Transatomic Power also belongs to this type. It was moderated by zirconium hydrate (ZrH) in direct contact with the fuel salt and possibly chemically stabilized with lithium hydride (LiH) and yttrium hydride (YH) [137].

4.9.1.2. *Liquid moderator heterogeneous MSRs*

Moderators containing hydrogen or deuterium, based on liquid organic compounds, were also suggested during the Nuclear Energy for the Propulsion of Aircraft (NEPA) project, preceding the ANP program. However, they were found to have insufficient radiation stability, especially at the high-power densities required for nuclear propulsion, lasting for only 20 min. of operation at 1 kW/cm^3 [81].

The ANP program also identified hydrogen gas as too diffuse for use as a moderator, and the cooling of liquid hydrogen moderator as unfeasible in reactors operating at high temperature [81].

After the efforts on suspensions of uranium particles in molten hydroxides was dropped in 1950, work continued on fluoride salt fuels moderated by molten (^7Li , Na, K, Be, and Rb) hydroxides, deuterioxides, and hydroxide-fluoride mixtures. These moderators were separated from the fuel salts by metal or ceramic cladding [141].

Much of the work in the early 1950s focused on determining the corrosiveness of fluorides and hydroxides, with the latter being the more corrosive of the two. At the time, the nature of hydroxides corrosivity, correctly called oxoacidity, was poorly understood. Experiments were conducted to see whether corrosion caused by hydroxides could be reduced by additives. Trace

amounts of H₂O were shown to be beneficial while larger amounts appeared to increase corrosion [13]. Adding sodium carbonate to sodium hydroxide seemed to have little effect [136]. Adding sodium metal to sodium hydroxide seemed to increase corrosion [136, 142]. Studies of hydroxides continued throughout the ANP program; the tests hinted that corrosion mitigation was possible, however, the lack of a systematic approach hindered successful corrosion control and by 1955 the hydroxides were deemed too harsh on structural materials to permit use in high temperature reactors [4].

A few years later, the redox chemistry of non-aqueous systems was systematically studied by Trémillon and co-workers, who demonstrated that alkaline hydroxides corrosivity can be controlled thermodynamically [142, 143], rehabilitating the use of this interesting family of liquid moderators. In 2018, the use of hydroxides for moderation of thermal and epi-thermal reactors was patented [144].

Light water and heavy water were given high consideration during the ANP program as liquid moderators for concepts with circulating salt fuel, though most research went into concepts based on light water because of the smaller core sizes achievable. Concepts were considered using unpressurized and pressurized light water as moderator, which was thermally insulated from the fuel salt, as well as concepts using supercritical water as coolant and moderator [81].

An unpressurized and thermally insulated reactor that is moderated and reflected by light water was estimated to transfer 8-15% of reactor power to the water, and the low temperature difference to ambient air would entail prohibitively large heat exchangers for nuclear powered flight [81, 141]. Pressurized water concepts also entailed pressurizing the salt, resulting in increased challenges to pump shaft seals and heavy reactor pressure vessels [81]. Several hybrid MSR designs mostly Be reflected and with a solid or liquid moderator from the end of the ANP program are illustrated in Figure 8.

FIG. 8. Core concepts from ARE (Figures 51-58 and 64 from Ref. [81]).

The simple reactor core with two regions, cooled by liquid metal, and Be reflected has the disadvantage of needing higher fissile inventory and higher probability of stagnant fuel in the larger void, when compared to the core with three regions, and reflected and moderated by Be. The latter core also has a more uniform power distribution, but there were technical difficulties of cooling the Be island in the core centre by liquid metal [81]. Cooling of the Be reflector and moderator is alleviated by an intermediate cooling layer with liquid metal in the core with five regions. The heat dissipation problem favours designs with a liquid moderator, where the moderator can be circulated and cooled as seen in the designs that are hydroxide moderated and Be reflected [81]. Core designs that are beryllium reflected and graphite moderated were also considered, where the graphite with fuel channels or sheets is located at the centre of the core, and the graphite is cooled by direct contact with the salt. They showed slight improvement from a nuclear perspective than the design purely reflected by Be [81].

TABLE 6. FAMILY III. 5. - HETEROGENOUS MSRS WITH NON-GRAPHITE MODERATOR

Class	Family	Type	Concept name	Designer	Appear in year	Ref.
III.	5.	Solid moderator heterogeneous MSRs	ARE	ORNL	1954	[81]
			ART concept	ORNL	1954	[81]
			ART concept variation	ORNL	1954	[81]
		Liquid moderator heterogeneous MSRs	TAP	Transatomic Power	2013	[137]
			Copenhagen Atomics Waste Burner	Copenhagen Atomics	2015	[145]
			CMSR	Seaborg Technologies	2018	[144]
			HW-MSR	SINAP	2019	[146]

4.10. HETEROGENEOUS CHLORIDE FAST MSRS

The reactor concepts belonging to this family do not use any moderator and the neutron spectrum is fast. The fuel has the form of liquid actinide chlorides with or without some chloride carrier salts. The carrier salts can be NaCl, MgCl₂, KCl, CaCl₂, or a mixture of them. Other options for the carrier salts are possible but seldom reported. As a common practice for this family, the fuel salt does not perform the coolant function and is physically separated from the dedicated coolant. In some older concepts, the blanket salt was used as the fuel coolant [147, 148]. The options for the fuel cycle can be U-Pu burning or breeding, U-Th burning or breeding, or TRU burning, thus allowing the reactors in this family to be suitable for different applications. Due to the presence of structural materials and dedicated coolant, the neutron economy of Heterogeneous chloride fast MSRs is tighter when compared to Homogeneous chloride fast MSRs. Nevertheless, the once-trough breed-and-burn U-Pu cycle is possible [149, 150].

Reactors in this family share many characteristics with other fast spectrum MSRs. They allow the possibilities of waste incineration of the actinides that are long lived and breeding for resource sustainability. The installed core power depends on the heat exchange surface of the separating material. At the same time, liquid fuel can withstand higher temperatures than the solid fuel and the dedicated coolant circulation can be fast. Hence, compact reactor core designs with a high core power density are possible. The absence of moderator also eliminates one of the constraints on the reactor core lifetime set by moderator materials such as graphite. Nonetheless, the lifetime of fuel pins is also limited. Thanks to the reduced capture cross-section of the fission products (FPs) in the fast spectrum, there is no need for frequent salt treatment to remove neutron poisons. On the other hand, fast reactors have hard neutron spectra causing more severe radiation damage in terms of dpa to those structural materials that are in direct contact to the salt and not shielded by a moderator or reflector than thermal reactors.

Unlike the Homogenous MSRs, the reactors in this family usually use fuel pins or assemblies in the reactor core to contain fuel salts. The approach reduces the volume of the fuel salt. The arrangement of the core can be similar to the ones commonly seen in the reactors cooled by

water or the sodium fast reactors, so it may help to reduce the difficulty in modelling the reactor core. For reactors that do not circulate the fuel salt, the contamination of the components by the fuel salt can be minimized. However, the introduction of the structural materials into the reactor core will impact the neutron economy and require the qualification of the structures such as the fuel pins or assemblies. The fuel of Heterogeneous chloride fast MSR usually utilize enriched ^{37}Cl as the salt base.

The fuel salts employed by this family are usually mixtures of chlorides due to their favourable features in terms of melting and boiling points, low capture cross-sections, and high solubility of actinide and lanthanide chlorides. The high solubility of actinides such as plutonium is particularly important for this family since fast reactors require a much larger fissile and fertile inventory than their thermal counterparts. Chloride fuel is expected to have good stability at high burnup and at high temperatures. Chloride fuel also has good thermophysical properties with respect to other types of nuclear fuel in terms of density, heat capacity, and thermal conductivity.

With respect to other candidate salts for fast MSR (mainly fluoride salts), chloride salts have a superior solubility of actinides. Reactors based on the chloride fuel tend to have harder neutron spectra, which may result in better burning rates if designed as waste burners. Historically, more effort was devoted to developing MSR using fluoride salts than to those using chloride salts. The issues of using the FLiBe salt, which was the one used in the MSRE by ORNL, caused by the ^6Li isotope and the beryllium health hazard are widely recognized. On the other hand, the ^{36}Cl that is produced mainly from the activation of ^{35}Cl may be a challenge for the disposal of spent nuclear fuel from the reactors using the chloride fuel salt because ^{36}Cl has an extremely long half-life and a high solubility in water.

Different options for fuel cycles are available in this family. For TRU burning, the reactors are usually fuelled by spent nuclear fuel from reactors cooled by water. For U-Pu burning or breeding, the reactors are usually fuelled by uranium and reactor grade plutonium. For Th-U burning or breeding, the reactors are usually fuelled by thorium, reactor grade plutonium or uranium.

4.10.1. Major reactor types

Historically, the non-circulating salt fuel was considered in the ORNL ANP program but rejected for that particular application. In the 1960s a gas cooled reactor concept with molten chlorides fuel was proposed at ORNL [147]. Nonetheless, the major research was focusing on MSR moderated by graphite and not on Heterogeneous chloride fast MSR. The major reactor types in this family are described in the following text, and major concepts introduced in Table 7.

4.10.1.1. Heterogeneous salt cooled fast MSR

This type of MSR rely on molten chloride fuel, which is typically cooled by dedicated chloride salt coolant. In the 1970s several concepts of blanket salt cooled Molten Chloride Fast Breeder Reactor (MCFBR) was proposed at EIR [35, 148], where the coolant salt was also acting as fertile material. In recent history, fluoride salt cooled chloride fuel salt system was proposed in [151, 152]. The concept was later modified for chloride salt coolant and several purposes. The SSR-W300 is developed by Moltex Energy in Canada and the UK. It is a 300 MW(e) waste

burner using a mixture of reactor grade plutonium chlorides and depleted uranium chlorides as the fuel. The fuel salt is contained in the fuel pins and is not forced to circulate; therefore, the design is called a Stable Salt Reactor (SSR). The same design was used to assess the breed-and-burn capability of this concept [149, 150]. It is possible to operate SSR in this fuel cycle type, but the neutron economy can be tight.

4.10.1.2. Heterogeneous lead cooled fast MSRs

Other coolants have been considered for the liquid chloride salt encapsulated in fuel pins. As already mentioned, a gas cooled option was proposed in the 1960s [147]. From the other two typical fast reactor coolants, i.e. liquid metals, only lead was proposed in the literature so far. DFR series reactors, developed by Institute for Solid-State Nuclear Physics (Institut für Festkörper-Kernphysik, or IFK) in Berlin, in Germany. DFR stands for Dual Fluid Reactor because the fuel salt and the coolant are circulated via different loops [153]. Their power levels range from 130 to 1500 MW(e). They use uranium, plutonium, or thorium chlorides as the fuel without carrier salts and liquid lead as the coolant. Currently there is also the HSR 50 concept [154], developed by Aristos Power in France. HSR stands for hard spectrum reactor. It is a fast spectrum reactor using chloride salt fuel and liquid lead as coolant that generates 50 MW(th). So far, none of the concepts considers sodium as the coolant.

TABLE 7. FAMILY III. 6. - HETEROGENEOUS CHLORIDE FAST MSRS

Class	Family	Type	Concept name	Designer	Appear in year	Ref.
III.	6.	Heterogeneous salt cooled fast MSRs	SSR-W300	Moltex	2014	[151, 152]
			SSR-B&B	Kasam and Shwageraus	2016 - 2020	[149, 150]
			MCFBR	EIR	1972 - 1974	[35, 148]
		Heterogeneous lead cooled fast MSRs	DFR	IFK Berlin	2015	[153]
			HSR	Aristos power	2018	[154]

4.11. OTHER MSRS

There are several MSR concepts that do not fit the six major families. Even though these concepts have the lowest TRL level and are rather exotic, they are still classified herein into several types. Each is briefly described next.

4.11.1. Major reactor types

The major other reactor types are described in the following text, and major concepts introduced in Table 8.

4.11.1.1. *Directly cooled MSR*s

The Directly cooled MSR's were considered in the past. There are about half a dozen concepts ranging from 1955 to 1974. The ideal of directly cooled MSR's is to avoid the large heat exchange surface needed for the salts, because of their low thermal conductivity, which is similar to water. The first reactor in this family was proposed in 1955 using a uranium solution in bismuth as fuel and KCl-LiCl as salt coolant [155]. The more common approach of chloride fuel salt cooled by direct contact with lead was first proposed in 1966 [126]. A similar solution was also considered in Ref. [42]. The fluoride salt cooled by lead was analysed in Ref. [156], and a rather exotic concept of chloride fuel salt cooled by boiling AlCl_3 was proposed in Ref. [33]. The biggest disadvantage of the "Directly cooled MSR's" family is the coolant contamination by actinides and fission products.

4.11.1.2. *Subcritical MSR*s

The Subcritical MSR's were studied in the 1990s, when great interest was dedicated to the Accelerator Driven System (ADS). The purpose was solely the waste burning. There were concepts moderated by graphite with fluoride salt [157, 158], homogeneous fluoride MSR's moderated by a graphite reflector [159], and chloride fast MSR's [160]. The major disadvantage of Subcritical reactors is the cost of the accelerator. Its function is to serve as a sophisticated control rod, which is compensating for the insufficiency in thermal feedback coefficients of some ADSs. Since an MSR using liquid fuel can be designed as a waste burner with all thermal reactivity feedback coefficients negative, the accelerator becomes unnecessary in concepts like MOSART.

4.11.1.3. *Hybrid moderator MSR*s

Many of the reactor concepts considered by the ANP program were hybrids of solid and liquid moderators. Solid metal Be moderator performance was deemed the best. Reflector materials were treated similarly, and also NaOH, NaOD, and $^7\text{LiOD}$ were considered. However, their moderating performance was limited by the predominantly forward scattering, and Be performed better because it is heavier [81]. The Hybrid moderator MSR's utilize a homogeneous mixture of moderator and fuel salt: this could be a mixture of liquid moderator and fuel salt, for example, hydroxide-fluorides mixtures, or principally a mixture of fuel salt and moderator slurry with solid particles suspended in fuel salt. The ANP program was aimed at developing a supersonic aircraft that was nuclear powered, and focused development on light and compact reactor designs that operated at high temperature, and shielding that was light weight. Early concepts included solid fuel reactors, liquid metal-alloy reactors, and reactors with boiling coolant and solid fuel that were cooled and moderated by sodium hydroxide [139, 161]. By 1950, the ANP program had narrowed in on the circulating fuel systems, when E. Bettis and R. C. Briant proposed molten fluoride salt systems, or fused salts, as they were referred to initially [162]. Concepts early in 1950 considered NaF-BeF₂-UF₄ as a self-moderating circulating fuel [89], and suspensions of uranium compounds in molten sodium hydroxide, including uranium chloride and fluoride [136]. However, the latter concepts were dropped by the summer of 1950 due to lack of solubility to actinide compounds and reactions between hydroxide and fluoride species [4].

The reflector moderated MSR's also can be included here. They are on the edge between Graphite moderated MSR's and Homogeneous fluoride fast MSR's. The fluoride salt includes

beryllium and has thus itself moderation power. The homogeneous core is, furthermore, surrounded by graphite reflector. The importance of thermal spectrum decreases with increasing core size. Accordingly, some of the bigger cores surrounded by graphite, such as the first homogeneous fluoride fast MSR proposed at ORSORT in 1952 [9], have epithermal or fast spectra. Some of the Fireball concepts developed within the ANP program belonging to this category is discussed in more detail in the Non-graphite moderated MSRs subsection.

4.11.1.4. Chloride salt cooled fast reactors

The chloride salt can be also used as coolant for a fast reactor with solid fuel and a bonding material in the fuel pin. The boiling AlCl_3 was considered as a coolant in Ref. [34]. The disadvantages of the usual coolants for fast reactors can be avoided by using chloride salt as a coolant. However, the salt coolant introduces additional complexity. There is also a more recent evaluation of molten salt as the coolant of fast reactors in concepts with a liquid salt-cooled reactor with flexible conversion ratio (LSFR) [163].

4.11.1.5. Frozen salt reactors

The Frozen salt reactors rely on solid fuel in salt form. Accordingly, they belong to Heterogeneous MSRs which require dedicated liquid coolant and separating structural material. The frozen state enables low operating temperatures. At the same time, the fuel can be occasionally re-melted and purified. Therefore, this concept has similar advantages and disadvantages as a reactor with solid fuel including orificing of the coolant flow and possible hotspots presence. The major advantage and disadvantage is the fuel melting temperature. It may be too low to be frozen during nominal operation and too high to be re-melted for purification. The Shanghai Institute of Applied Physics built a frozen salt zero-power test reactor in 1971 [164].

4.11.1.6. Hybrid spectrum MSRs

Hybrid spectrum MSRs are characterized by the presence of both thermal and fast spectra in the core. This design option is usually applied to minimise the neutron leakage. The core often acts as a flux trap, where the multiplication factor is much higher in the centre than on the periphery [165]. There are also concepts with variable neutron spectra, where manipulation with the core composition or geometry results in spectral changes [166].

4.11.1.7. Heterogeneous gas cooled MSRs

The gas coolant is considered for Heterogeneous chloride fast MSR in one paper by Alexander from 1963 [147].

TABLE 8. CLASS IV - OTHER MSRS

Class	Type	Concept	Designer	Appear in year	Ref.
IV.	Directly cooled MSRs	U solution in molten bismuth cooled by KCl-LiCl salt	Burch et al.	1955	[155]
		Chloride fuel salt cooled by lead	Moore and Fawcett	1966	[126]
		Chloride fuel salt cooled by boiling AlCl ₃	Taube et. al.	1967	[33]
		Fluoride fuel salt cooled by lead	Gat	1967	[156]
		Chloride fuel salt cooled by lead	Smith et al.	1974	[42]
	Subcritical MSRs	ATW	Bowman	1997	[158, 159]
		JAERI OMEGA	Katsuta	1997	[160]
		SPHINX	Hron	2000	[157]
	Hybrid moderator MSRs	Moderator diluted in the salt	Ellis and Thompson	1950	[89]
		Reflector moderated homogeneous MSR	Wehmeyer	1953	[9]
		ART concept variation	ORNL	1954	[81]
	Chloride salt cooled fast reactors	Carbide-Fuelled Fast Breeder Reactor	Taube	1969	[34]
		LSFR	Todreas et al.	2009	[163]
	Frozen salt reactors	Zero-power (cold) MSR	Shanghai Institute of Applied Physics	1971	[164]
	Hybrid spectrum MSRs	Flux trap reactor	Krepel et al.	2014	[165]
		Variable spectrum reactor	Transatomic	2015	[166]
	Heterogeneous gas cooled MSRs	Chloride salt fuel cooled by gas	Alexander	1963	[147]

5. RESEARCH AND DEVELOPMENT ACTIVITIES

5.1. INTRODUCTION

This section presents the research and development (R&D) activities in Member States with major programmes in the public or private sector, or both. The following subsections are sorted by Member State's name in alphabetical order. One Member State known to have major developments in MSR technology is India; unfortunately, its input was not received for inclusion in this document.

5.2. R&D ACTIVITIES IN CANADA

5.2.1. Background

Canada's nuclear sector is currently focused on Small Modular Reactors (SMRs) that have the potential to provide clean energy for applications ranging from remote communities, to heavy industry, to electricity generation at grid scale.⁴ In November 2018, the federal government released the Canadian SMR Roadmap that outlines recommendations for collaboration between federal, provincial and territorial governments and other stakeholders to support SMR development in Canada.⁵ In February 2020, the government of Canada announced that the launch of Canada's SMR Action Plan will be in November 2020.⁶ At the provincial level, Ontario, Saskatchewan and New Brunswick signed a memorandum of understanding (MOU) in December 2019, and were joined in August 2020 by Alberta, to work together on the development and deployment of SMRs in Canada, including technological readiness, regulatory frameworks, and nuclear waste management.

Many SMR developers have initiated a pre-licensing Vendor Design Review (VDR) with the Canadian Nuclear Safety Commission (CNSC).⁷ Two small modular MSRs are among a range of SMR designs under the pre-licensing VDR: Terrestrial Energy and Moltex Energy. Terrestrial Energy is developing an MSR reactor with fluoride fuel salt, thermal spectrum (graphite moderator), and integral heat exchangers named the Integral MSR (IMSR). Moltex Energy is developing an MSR with chloride fuel salt and fast spectrum named Stable Salt Reactor (SSR) that is tailored to burn spent fuel from CANDU (CANadian Deuterium Uranium) plants.

In parallel, experienced Canadian operators of generation of nuclear power are working with SMR developers to advance engineering and design work and evaluate potential demonstration projects. IMSR by Terrestrial is one of the SMR designs selected by Ontario Power Generation (OPG) and SSR by Moltex is one of the SMR designs affiliated with New Brunswick (NB) Power. As part of these efforts, the Canadian government announced a Strategic Innovation Fund (SIF) investment of 20 million in October 2020 that will assist Terrestrial Energy to

⁴ GOVERNMENT OF CANADA, Canada's Mid-Century Long-Term Low-Greenhouse Gas Development Strategy (2020), https://unfccc.int/files/focus/long-term_strategies/application/pdf/canadas_mid-century_long-term_strategy.pdf

⁵ GOVERNMENT OF CANADA, Canadian SMR Roadmap (2020), <https://smrroadmap.ca/>

⁶ GOVERNMENT OF CANADA, Canada's Small Modular Reactor Action Plan (2020), <https://www.nrcan.gc.ca/our-natural-resources/energy-sources-distribution/nuclear-energy-uranium/canadas-small-modular-reactor-action-plan/21183>

⁷ CANADIAN NUCLEAR SAFETY COMMISSION, Pre-Licensing Vendor Design Review (2020), <https://www.nuclearsafety.gc.ca/eng/reactors/power-plants/pre-licensing-vendor-design-review/index.cfm>

complete a key pre-licensing milestone through the CNSC.⁸ The SIF programme is designed to attract and support business investments of high quality across all sectors of the Canadian economy.

5.2.2. R&D activities

Molten salt technology is a new area of nuclear research and development in Canada. Canada's premier nuclear Science & Technology (S&T) organization is Canadian Nuclear Laboratories (CNL). To support the government's areas of focus, CNL has identified SMRs as one of its key strategic initiatives. As part of this programme, CNL is building its expertise and capabilities to support the development and deployment of different SMR technologies. In particular, several MSR research projects were funded by the government of Canada through Atomic Energy of Canada Limited (AECL), and launched in 2016 at CNL's Chalk River Laboratories (CRL) site. CRL site is Canada's largest S&T complex. Historically, the expertise and research facilities at CRL were used to develop the CANDU technology. These capabilities are expanding, and new capabilities are being developed, towards qualification and demonstration of molten salt technology among other SMR advanced technologies. At present, CNL's MSR R&D programs address the following areas:

- Development of experimental protocols, integrated with quality assurance plans, for verification and characterization of key transport and thermodynamic properties of fuel and coolant salts to support safety assessment and licensing of MSR technology;
- Development of experimental facilities for synthesis of actinide fluorides and chlorides;
- Experimental program to investigate molten salt's fuel behaviour in accident conditions, including the measurement of release of fission products from molten salt fuels;
- Molecular Dynamics (MD) for predicting thermophysical properties of fuel and coolant salts;
- Corrosion of materials that are in contact with fuel and coolant salts;
- Passive decay heat removal in MSRs;
- Development of simulation tools with coupled Computational Fluid Dynamics (CFD)-neutronics for MSR transients;
- Development of modelling capabilities for beyond design basis events for MSRs;
- Phenomenon identification and ranking table (PIRT) analysis for severe accidents in MSRs;
- Evaluation of waste streams from MSRs considered for deployment in Canada;
- Approaches and techniques for addressing safeguards and non-proliferation issues of MSRs.

Another important initiative is CNL's Canadian Nuclear Research Initiative (CNRI) that was launched in June 2019 to enable SMR technology developers to gain access to CNL's expertise and facilities on a cost share basis.⁹ The objective of the CNRI programme is to optimize resources and enable collaborative research work with the SMR industry to accelerate the deployment of SMRs in Canada that are safe, reliable, and cost effective. To date, the MSR research projects funded by CNRI are:

⁸ GOVERNMENT OF CANADA, Government of Canada Invests in Innovative Small Modular Reactor Technology (2020), <https://www.canada.ca/en/innovation-science-economic-development/news/2020/10/government-of-canada-invests-in-innovative-small-modular-reactor-technology.html>

⁹ CANADIAN NUCLEAR LABORATORIES, CNL's Canadian Nuclear Research Initiative (2020), <https://www.cnl.ca/en/home/commercial/cnl-s-canadian-nuclear-research-initiative.aspx>

- Research and engineering of technologies to better separate, analyse, and store tritium generated through the operation of Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR) design.¹⁰
- Support Moltex Energy to demonstrate the commercial viability of Waste To Stable Salts (WATSS) technology to convert spent CANDU fuel into new fuel that can power Stable Salt Reactors (SSR).¹¹
- Evaluation of safeguards related to the operation of Terrestrial's IMSR.¹²

In addition to CNL, MSR R&D activities are also performed by the following universities:

- The Polytechnique Montréal: A technological university that maintains and develops thermodynamic model databases for salts/oxides systems in the FactSage [167] thermochemical software, which is licensed to universities and companies around the world. Currently, one of the thermodynamic models under development is Kairos Power's thermodynamic/phase equilibrium modelling of molten 2LiF-BeF₂ (FLiBe) salt with corrosion products.
- Ontario Tech University: Several research projects related to fluoride salts are underway. Capabilities are being developed to model molten salts with the open source code Thermochemica [168], which is being coupled to various multiphysics codes. In particular, a new code called Yellowjacket is under development in partnership with Idaho National Laboratory (INL) to simulate lower-length scale corrosion by fluoride salts. In this context, 'lower-length scale' refers to the direct coupling of phase field calculations of microstructure evolution and computational thermodynamics.
- University of New Brunswick (UNB): UNB's Centre for Nuclear Energy Research (CNER) is a partner in NB's nuclear research cluster and is focusing on developing experimental capabilities in MSR R&D related to Moltex SSR technology. Moltex Energy is one of the two developers of SMRs with fast spectrum who committed \$5 million to match funds from NB's government, through New Brunswick Energy Solutions Corporation to support MSR R&D activities. This corporation is a joint venture formed in May 2017 by NB's provincial government and NB Power, operator of the Point Lepreau nuclear power plant.¹³

At the international level, Canada is considering joining the Generation IV International Forum (GIF) MSR system for international R&D collaborations, and is working with other GIF members to integrate an SMR focus into the GIF programme. The GIF MSR is currently operating under a memorandum of understanding (MOU), and is in the process of transitioning to a System Arrangement. With support from Natural Resources Canada (NRCan), Terrestrial Energy has joined the MSR MOU as a signatory in May 2019. NRCan supported Moltex Energy's participation in a meeting of the GIF MSR provisional System Steering Committee

¹⁰ CANADIAN NUCLEAR LABORATORIES, CNL Partners with Kairos Power on SMR Research (2020), <https://www.cnl.ca/en/home/news-and-publications/news-releases/2020/cnl-partners-with-kairos-power-on-smr-research.aspx>

¹¹ CANADIAN NUCLEAR LABORATORIES, CNL & Moltex Energy Partner on SMR Fuel Research (2020), <https://www.cnl.ca/en/home/news-and-publications/news-releases/2020/cnl-moltex-energy-partner-on-smr-fuel-research.aspx>

¹² CANADIAN NUCLEAR LABORATORIES, CNL & Terrestrial Energy Partner on SMR Fuel Research (2020), <https://www.cnl.ca/en/home/news-and-publications/news-releases/2020/cnl-terrestrial-energy-partner-on-smr-fuel-research.aspx>

¹³ Government of New Brunswick, \$10 Million Committed for Nuclear Research Cluster (2018), https://www2.gnb.ca/content/gnb/en/news/news_release.2018.06.0832.html

(pSSC) that was held in September 2019. CNL is currently an observer in this pSSC and the project Severe Accident Modelling and Safety Assessment for Fluid-fuel Energy Reactors (SAMOSAFER).

5.3. R&D ACTIVITIES IN CHINA

SINAP, Chinese Academy of Sciences (CAS) restarted a research programme on thorium utilization named TMSR (Thorium Molten Salt Reactor) in January 2011. This programme focused on the research and development of technologies related to the TMSR, to achieve safe and economic use of Generation IV MSR NPPs and effective use of the thorium resource. The TMSR program developed a TMSR simulator (mock-up) and test reactor to prove the concept, then plans to develop the demonstration reactors to validate TMSR technologies, and finally the commercial NPPs will be deployed. SINAP built the TMSR simulator (mock-up) in 2019, the construction of a 2 MW(th) TMSR test reactor will be complete in 2021, and a 100 MW(e) small modular TMSR demonstration reactor is expected to be deployed before 2030.

5.3.1. R&D progress of TMSR technology

The TMSR program already achieved several results in key TMSR technologies. Some of the main achievements are presented in the subsections below [169].

5.3.1.1. Molten salts

Fluoride molten salts (in this subsection referred to as salts for brevity) are important materials in the TMSR. FLiBe is used as coolant in the primary loop, and FLiNaK is used as coolant in the secondary loop. In the liquid fuel TMSR, thorium and uranium are dissolved in the FLiBe. The TMSR team developed advanced techniques for the preparation and purification of these salts, and it has the capabilities to test and evaluate them.

The corrosion rates of structural materials in contact with the salts are important for safely using the salts in the TMSR. It is well known that the corrosion rate is greatly dependent on impurities, such as oxygen, water and sulfur, in the salts. Consequently, the impurities must be reduced to an appropriate level before the salts can be used in the reactor's loops. The TMSR team has extensive experience in controlling the impurities.

The specifications of the FLiBe to be used in the primary loop are strictly controlled. The two main requirements for the FLiBe are impurities control and abundance of lithium isotope. The isotopic abundance of ${}^7\text{Li}$ for the preparation of the salt must be over 99.99%. A new centrifugal extraction technique was developed as an alternative to the traditional amalgam method. The new extraction technique is much more environmentally friendly than the amalgam method since no mercury is used during salt production.

Beryllium fluoride (BeF_2) is another important raw material for the production of FLiBe. Before purification treatment, beryllium fluoride contains 2000 ppm sulfur and 30 ppm boron. However, after desulfurization at high temperature and recrystallization with ammonium, all the impurities are significantly reduced: the sulfur and phosphorus concentrations are below 20 ppm, and the nitrogen and boron concentrations are below 1.5 ppm. More importantly, the TMSR team successfully combined all these new techniques to produce nuclear grade FLiBe

to be used in the both solid fuel and liquid fuel TMSR designs. Figure 9 shows fluoride salts and their production facilities.

FIG. 9. Fluoride salts and production facilities.

The main requirement for the FLiNaK salt to be used in the secondary loop is impurities control. The TMSR team found that if the concentration of oxygen is around 100 ppm, the corrosion rate of Hastelloy-N alloy (UNS N10003) is about 20 $\mu\text{m}/\text{year}$. It is desirable to reduce the corrosion rate by controlling the impurities in the salt. The team used an ammonium hydrogen fluoride method to remove the oxygen. A ton of FLiNaK salt was produced by using this method and was tested in an experimental loop over a year without any problems. To reduce the impurities further, a hydrofluoric acid-hydrogen process (HF-H₂ process) has been used. The quality of the salt prepared by this process is dependent on the impurities in the hydrofluoric acid. The team developed a new technique to purify commercial hydrofluoric acid. The corrosion rate of Hastelloy-N alloy (UNS N10003) in FLiNaK salt prepared by this technique is under 2 $\mu\text{m}/\text{year}$.

5.3.1.2. Alloy based on nickel and graphite

The structural materials of the TMSR will be subjected to extreme environments, that is, high temperature, high neutron fluence, and a corrosive coolant. In the case of the liquid fuel design, the dissolved fuel in the fluoride salt will produce a few radioactive and corrosive substances (such as Xe, F, I) under neutron irradiation in the core. Hence, the structural materials must have good strength at high temperature, good resistance to corrosion, and high tolerance to neutron irradiation. An alloy based on nickel and graphite are the two main structural materials to be used in the TMSR.

The alloy based on nickel is considered to be the primary option for metallic structural materials. An alloy based on nickel, named GH3535, was developed and its performance parameters are comparable with those of Hastelloy-N produced in the US. Production of GH3535 alloy at pilot scale was completed and a technique for mass production was established. Furthermore, the TMSR team developed several processing methods for alloys based on nickel, such as the hot extrusion and rolling process for pipe of large calibre to be used in the reactor loop, as well as tungsten inert gas welding for thick plates that meets ASME requirements. These key technologies ensure the successful application of GH3535 alloy in reactor components like pressure vessels, loops and, control rod sleeves.

Due to high thermal conductivity and low chemical activity at elevated temperatures, graphite is considered an ideal moderator for TMSR, as well as a structural material. The TMSR team improved a fabrication method for isostatic graphite. The new method has been applied to produce a superfine grain graphite, NG-CT-50, for the liquid fuel TMSR. The superfine grain graphite has high density and other excellent material properties such as high bending and compressive strengths, low porosity, and low boron. A series of compatibility tests of graphite with molten salt were conducted to assess the effect of molten salt impregnation.

5.3.1.3. *Corrosion experiment*

The TMSR team studied the corrosion of various structure materials in fluoride salts. The results were used to optimize the composition of the alloys to mitigate the corrosion. Thermal diffusion and bi-metallic methods were also tested. The corrosion of metallic samples in fluoride salts of different purity was studied, and it was found that maintaining the oxygen level below 100 ppm in FLiNaK salt can significantly reduce the corrosion. The team also studied the infiltration of fluoride salts into graphite, and demonstrated that the ultrafine-grain graphite to be used in TMSR test reactors has excellent resistance against salt infiltration. The team constructed a natural circulation loop to study the dynamic corrosion by the fluoride salts. The dynamic corrosion experiment was completed in 2018.

5.3.1.4. *Irradiation of thorium, alloys, and graphite*

MSRs are often associated with the Th-U fuel cycle [170]. The TMSR team irradiated thorium, alloys, and graphite samples to obtain irradiation data. The team irradiated 9.7 g of thorium oxides in 2015 to use the irradiated sample to conduct basic research on the separation of thorium and uranium. The number of neutrons accumulated on the sample during the irradiation is in the order of 10^{19} per cm^2 . The team also irradiated thorium fluorides with and without other fluoride salt mixtures to study the behaviour of the thorium fluoride fuel in 2018. It also irradiated nickel alloy and nuclear graphite samples to 2.5×10^{19} and 5×10^{20} neutrons per cm^2 , respectively, at temperatures of 650 and 700°C. These are the design operating conditions of the test reactors. The irradiations were completed, and the samples are being subjected to post irradiation examinations.

5.3.1.5. *Components and instruments*

The TMSR team designed and built prototypical components and instruments for the test reactors. The prototypical fluoride salt pump was manufactured using a nickel alloy. The design operating temperature is 700°C, the flow rate is 300 m^3/h , and the pump head is 20 meters. The characteristics of the pump meet the requirements of the test reactors. The team also developed freezing valves to be used in the test reactors, and designed other major components such as salt-salt heat exchangers, salt-air heat exchangers, reactor vessels, in-vessel structures, and control rods. Prototypes were produced by manufacturers and are currently undergoing tests in the design operating conditions.

The team also developed instruments for high temperature operation, such as flowmeter, manometer, thermometer, and tank gauge that can be used in the fluoride salt test reactors. It built equipment that can be used to calibrate these instruments at temperatures over 700°C. It also developed software that can be used to control the test reactors and to simulate the behaviour of the reactors under different conditions.

5.3.1.6. *Test loops*

The team constructed several test loops that can create an experimental complex to support future MSR R&D. These loops use fluoride salt and other kinds of surrogate fluids. The fluoride salt test loop uses FLiNaK salt and operates with a flow rate of 15 m^3/h at a temperature of 650°C. The loop was mainly used for testing components and obtaining fluoride salt operating experience. A natural circulation experimental loop using nitrates salt as the

surrogate fluid is able to conduct experiments of the direct reactor auxiliary cooling system. The loop is capable of removing 20 kilowatts of heat just by natural circulation of the salt and the air, a phenomenon that can be relied upon to passively remove the reactor's decay heat when all power is lost. Another nitrates salt loop with forced circulation was used for thermo hydraulic experiments. A specially designed Dowtherm-A oil loop can be used to perform experiments to simulate some fluoride salt experiments when scaling analysis is correctly performed. The advantage of using oil as the surrogate fluid to replace fluoride salts is the significant cost reduction. There are also several water loops for various purposes, such as the hydraulics test of salt pumps, the visualization of the flow field, and the performance test of the fluid diode. Figure 10 shows pictures of pumps, heat exchangers, and loops.

FIG. 10. Key equipment and molten salt loops developed in SINAR.

High Temperature molten salt (HTS) thermohydraulic experiment loop. This loop is the first molten salt experiment loop at high temperature of TMSR. The HTS molten salt thermohydraulic experiment loop system scheme is shown in Figure 11, which can be divided into five subsystems: a) Molten salt circulation system; b) Molten salt storage system; c) Heat conduction test system; d) Pneumatic system; e) Control and safety interlock system. Based on the this loop, various research will be carried out: (1) Research and experimental platforms for key HTS molten salt equipment; (2) Research on this loop's thermohydraulic characteristics to provide basis for the design, construction and operation of a TMSR loop; (3) Research on the molten salt loop control and safety interlock system for the safe and reliable operation of TMSR; and (4) Research on molten salt loop operation technology, and establishing the database for the loop's operating parameters provides a basis for developing the operating procedures for the TMSR loop.

FIG. 11. Overall layout of HTS molten salt thermal experiment loop.

FliNaK molten salt high temperature experiment loop [171]. The design and construction of this loop aims to study the resistance at high temperature and fluoride salt resistance of this loop and related equipment; carry out research on molten salt flow and heat transfer characteristics; provide a research platform for developing and testing key equipment; perform the following tests: Gas simulation test, fluoride salt deoxygenation purification test, thermal test, study the control and safety interlock; and improve safety. This loop is shown in Figure 12. It is mainly composed of molten salt circulation system, molten salt storage and transfer system, gas circuit system, heat tracing and insulation system, measurement system, control and safety interlocking system, pretreatment system, and other key components. It can provide design basis and information for the design, construction, operation and safety control of the molten salt loop system of a reactor. This loop uses FliNaK salt as the working medium. The fluorine salt is a eutectic composite salt composed of LiF-NaF-KF with a melting point of 454°C and a boiling point of 1570°C, which is commonly used to simulate molten salt for molten salt reactor loop systems and as a test bench.

FIG. 12. Overall layout of the FLiNaK molten salt high temperature experiment loop.

Nitrate natural circulation experiment (NNCL) loop. This system, shown in Figure 13, mainly includes (1) main circuit system; (2) air cooling tower system; (3) salt charging and discharging system; (4) measurement control system; (5) heat tracing insulation system. The main circuit system is the key system of this loop, and its function is to perform the natural circulation of molten salt. Its main components are DRACS heat exchanger (DHX), draft heat exchanger (NDHX), pipelines, valves, expansion tanks, heating cables, heaters, nitrates, and molten salt pools [172]. The cooling tower system is used to enhance the heat dissipation of the NDHX heat exchanger, and is mainly composed of ventilation tower, air door, heater, fan, lifting mechanism, positioning mechanism, and temperature and humidity measuring instrument. The salt charging and discharging system mainly includes gas system and vacuum system. The measurement control system is used to measure and control the temperature, liquid level, flow, and pressure of the loop system, and monitor system failures to ensure safe and reliable operation of the loop system. It is mainly composed of thermocouples, temperature controllers, molten salt flow meters, air flow meters, liquid level gauges, pressure sensors, data acquisition instruments, and other equipment. The heat tracing insulation system is used to heat the main circuit before molten salt is injected into the pipeline to prevent the molten salt from solidifying after entering the pipeline and causing blockage of the pipeline. It is mainly composed of heating plates, resistance wires, and circuits.

FIG. 13. Overall layout of NNCL loop.

5.3.1.7. Radiochemistry

In the thorium fuel cycle designed by TMSR, fissile materials including ^{233}U will be separated from the irradiated fuel and reloaded to the reactor. After being decontaminated, fluoride salt will also be reloaded to the reactor due to its high cost. TMSR chemists conducted research on processing of the irradiated fuel. Technologies such as fluorination for uranium recovery, distillation for salt purification, and fluorides electrochemical separation for uranium recovery were developed and tested in “cold” experiments [173]. The distillation device at lab scale can collect fluoride salt at a speed of 6 kg/h, and the decontamination factor (mass fraction ratio of analysed element in the evaporated sample and condensed salt) is greater than 100 for most neutron poisons. The separation ratio of the electrochemical test setup for uranium is over 99 percent. As shown in Figure 14, TMSR has already constructed hot cells for future “hot” experiments of the irradiated fuel, and they were put into operation in 2018. TMSR procured advanced instruments and developed techniques to accurately measure properties and compositions of the salts. TMSR labs can measure physical properties such as viscosity, density, and thermal conductivity. The labs are also capable of analysing trace elements, such as oxygen, boron, cadmium, and rare earth elements, in the salts.

FIG. 14. Radiochemistry facilities.

5.3.1.8. Tritium Management

TMSR chose FLiBe as the fuel salt for the test reactors. This salt generates tritium when irradiated by neutrons. TMSR has conducted research on tritium stripping using a bubbling system, tritium separation from noble gases using cryogenics, tritium storage using alloys, and continuous tritium monitoring to prevent an uncontrolled release of tritium into the environment [174]. The experiments using surrogate gases such as hydrogen and deuterium were carried out to evaluate the performance of different techniques. Some experiments were carried out on Pd/Ag alloy membrane diffuser, tritium storage alloy bed, and oxidation post-treatment bed using hydrogen isotope to simulate tritium control performance.

5.3.2. Design and construction of TMSR-0

SINAP designed and constructed a mock-up reactor, called TMSR-0, which provides data to validate the thermohydraulic simulation and computer codes for analysing the safety of the TMSR test reactor, and is a training platform for TMSR construction, operation, and maintenance.

The TMSR-0 reactor is a simulator of TMSR-SF1 (10 MW(e)) with scaled size of 1:3. The heat source of TMSR-0 used an electric heater with the maximum power of 400 kW. TMSR-0 is composed of reactor body, primary loop and secondary loop (with molten salt to air heat exchanger), passive residual decay heat removal system (DHRS) for natural gas circulation, and instrumentation and control systems. The layout of TMSR-0 is shown in Figure 15, and Table 9 gives TMSR-0's major technical parameters.

FIG. 15. Schematic layout of TMSR-0.

TABLE 9. MAJOR TECHNICAL PARAMETERS OF TMSR-0

Parameter	Value
Electric heater power (kW)	0-400
Coolant	FLiNaK
Mock-up core inlet/outlet coolant temperature (°C)	600/650
Air heat exchanger (IHX) inlet/outlet molten salt temperature (°C)	520/536
Air heat exchanger (IHX) inlet/outlet air temperature (°C)	40/180
Molten salt rated flow of primary circuit (kg/s)	0-10.0
Molten salt rated flow of secondary circuit (kg/s)	0-12.2
Power of the passive residual DHRS (kw)	12.8

FLiNaK molten salt is used for the primary and secondary loops of TMSR-0. The alloy based on Ni (GH3535) developed by the TMSR team is selected as the structural material for the reactor body and both loops of TMSR-0. The inlet and outlet temperatures of the primary loop are 600°C and 650°C, respectively. TMSR-0 can be used to do the verification experiments related to the design, safety, and components and instruments of TMSR-LF1.

From the perspective of developing the TMSR program for the long term, technical information from TMSR-0 can provide support for the design of molten salt experimental reactor and demonstration reactor, both based on thorium, and accelerate the development of the Generation IV MSR reactor.

The construction objectives of TMSR-0 include:

- Provide an overall engineering verification platform for the key technologies and equipment of TMSR-SF1 and TMSR-LF1 experimental reactors, and facilitate gaining the necessary experience for design, construction, operation, and maintenance of these reactors;
- Verify the thermohydraulic design and safety of TMSR-SF1 and TMSR-LF1 experimental reactors, verify the applicability of the design and analysis program, and support obtaining licenses for these reactors;
- Provide an important experimental platform for the future development of reactors cooled by molten salt. For example, this platform supports R&D of procedures for thermohydraulic and safety analyses and developing key technologies and equipment.

FIG. 16. Layout of TMSR-0.

The design and construction period of TMSR-0 was from April 2016 to March 2020. The layout is shown in Figure 17. The project design was completed in July 2016, the arrival and

acceptance of all equipment in September 2018, the construction and installation in 2019, and TMSR-0 started operation in March 2020.

5.3.3. Design and construction of the experimental reactor TMSR-LF1

5.3.3.1. Introduction

TMSR-LF1 is a molten salt experimental reactor with liquid fuel and a power design of 2 MW(th), and it employs an integrated layout that pump, heat exchanger are located in the reactor vessel. The technical goal of TMSR-LF1 is to develop the molten salt reactor based on liquid fuel (thorium) and verify its feasibility. TMSR-LF1 can provide technical support and experience for the design and construction of a small modular molten salt demonstration reactor based on thorium. Table 10 gives TMSR-LF1's main parameters, and Figure 17 presents a schematic layout of the TMSR-LF1's reactor vessel and core.

TABLE 10. MAIN PARAMETERS OF TMSR-LF1

Parameter	Value
Reactor type	MSR with liquid fuel
Thermal capacity, MW(th)	2
Design life (years)	10
Fuel	LiF-BeF ₂ -ZrF ₄ -UF ₄ -(ThF ₄)*
Moderator	Graphite
Core inlet/outlet coolant temperature (°C)	630/650
Rated flow of fuel salt circuit (kg/s)	50
U-235 enrichment (wt%)	19.75
Fuel loading/unloading	Initial loading by gas pressure, online loading by capsule, and unloading by gas pressure
Reactivity control	Two shutdown systems
Control rod absorber material	B ₄ C

* Thorium will be added as experimental material in operation period.

FIG. 17. Schematic layout of TMSR-LF1's reactor vessel and core.

Verification activities include: (1) Verification of basic principles: neutronic characteristics, thermohydraulic characteristics, and reactor safety characteristics; and (2) Technical verification, such as verifying equipment and technology related to materials, fuel salt, molten salt, radiochemistry, and loop and instrument control system.

5.3.3.2. System composition

The TMSR-LF1 generates 2 MW(th), its moderator is graphite, and the integrated layout with forced circulation is adopted. The main systems include integrated reactor system, cooling salt circuit system, instrument and control system, and auxiliary systems (such as gas circuit system, fuel salt loading/unloading, sampling system, three-waste treatment system, and heat tracing and insulation system). Heat is generated when the fuel salt flows through the core and is released when it flows through the salt-salt heat exchanger. TMSR-LF1 is an integral reactor with the core, main pump, and salt-salt heat exchanger located in the reactor vessel.

The reactor system of TMSR-LF1 includes the main reactor vessel, the core moderated by graphite, the flow distribution device, the control rod system, the relevant functional channels, the salt-salt heat exchanger, and the fuel salt circulation pump. UNS N10003 alloy is used for the metal components in contact with molten salt since it is resistant to corrosion.

Fuel salt. The nuclear fuel in TMSR-LF1 is a liquid fuel salt and operates at high temperature and at near atmospheric pressure. Liquid fuel salt needs to have excellent neutron properties, radiation resistance, high specific heat capacity, good fluidity and thermal conductivity, a wide range of working temperatures, and chemical stability. In addition, it must have good compatibility with materials, and a certain solubility for fuel (uranium and thorium) and fission products.

Reactor core. The core of TMSR-LF1 is composed of graphite components, on which there are fuel salt passages and functional channels. There are 244 fuel salt passages with a diameter of 40mm. The reactor core contains 10 functional channels with UNS N10003 alloy casing, including 6 control rod channels, 1 experimental measurement channel, 1 neutron source channel, 1 fuel loading/unloading channel, and 1 temperature measurement channel.

TMSR-LF1 has a total of 6 control rods, which belong to two independent systems. The reactivity control system includes 2 compensation rods, 1 regulating rod, and 1 safety rod. The shutdown system consists of two safety rods, which have a driving mechanism and rod shape that are different from those of the first shutdown system. All control rod absorbers are made of boron carbide (B_4C). The neutron source is an Am-Be source.

The main reactor vessel is mainly composed of a cylinder assembly and a roof assembly. The cylinder assembly mainly consists of flange, straight cylinder body, lower head, and inlet and outlet pipe of cooling salt loop. The flange and straight tube are made of UNS N10003 ring rolled pieces. The flange is the supporting part of the whole reactor. The straight cylinder bodies composed of a barrel stiffener and a barrel. The roof assembly is mainly composed of the top head and the upper channels socket. The top head is a butterfly head, welded with the pipe seat of the related channels.

Reactivity control systems. As mentioned above, a reactivity control system and a shutdown system are used to meet the core reactivity control and shutdown requirements in all possible states. The 4 control rods of the reactivity control system perform reactivity control functions during normal operation, including power regulation, temperature compensation, xenon poison effect compensation, fuel consumption compensation, delayed neutron flow effect, operation backup, and compensation for reactivity changes caused by experimental samples.

Thermohydraulic design. The basic function of the reactor's thermohydraulic design is to ensure that the reactor's heat transfer capacity is matched with the reactor's heat generation capacity during normal operation. In addition, the design provides a set of reasonable thermal parameters such as loop pressure and temperature for the thermal power transfer system. Further, it can transfer decay heat from the reactor to the ultimate heat sink after shutdown, and ensures the integrity of multi-channel barriers, which are designed to limit the release of radioactive products during potential accidents.

The temperature of UNS N10003 alloy structural material on the fuel salt pressure boundary of the reactor is lower than 700°C under normal operating conditions. The maximum temperature is lower than 800°C under the predicted operating events and design basis accidents (DBAs). The requirement is that the maximum temperature of fuel salt in the core should be less than 1200°C for DBAs. The minimum temperature of fuel salt in the reactor is above 500°C under normal operating conditions and predicted operating event conditions.

Fuel Management System. The fuel management system is used for initial fuel loading, and fuel salt discharge under normal or accident conditions. The fuel salt in these cases is driven by gas pressure and gravity.

Online Fission Gas Removal System. Fission gases and some noble metals are removed from the reactor by entrained cover gas. Then they flow into the off-gas process system for decay, filtration, and finally are released into the atmosphere.

5.3.3.3. *Safety features*

Reactivity safety. Changes in the temperature of a nuclear reactor leads to the changes in the reactor's reactivity, which is characterized by the temperature reactivity coefficient (TRC). The TMSR-LF1's TRC is strongly negative, which supports the control and safety of the reactor at various power levels.

As described above, the TMSR-LF1 has two different control rod systems, which will automatically operate in accidents. Low excess reactivity makes it possible to shut down the reactor by the effect of the very negative TRC. After the shutdown, the fuel salt can be drained from the reactor for shutdown for the long term.

Decay Heat Removal System. TMSR-LF1 has a passive DHRS. The verification test of this system's capability has been completed on the TMSR-0 simulation reactor, including verifying the reliability of the associated software.

Confinement System. The confinement system performs the safety function for containing radioactive nuclide after a hypothetical DBA. Therefore, the safety vessel of this system contains the fuel salt system and reactor.

5.3.3.4. *Plant safety and operational performance*

TMSR-LF1 design principles are the following: 1) effective process and design guidelines should be established and adopted, and 2) ensure that basic safety design principles are followed throughout the phases of design and design change.

5.3.3.5. *Project progress*

The TMSR-LF1 project is undertaken by the SINAP, CAS. Its site selection, environmental impact assessment, scheme design, preliminary engineering design and construction drawing design were completed.

In December 2018, the Chinese National Nuclear Safety Administration (NNSA) approved the TMSR-LF1 site selection for Hong Sha Gang Industrial Park in Wu Wei, Gansu Province. In July 2019, NNSA granted approval to the TMSR-LF1 design.

In August 2019, the technical review of the TMSR-LF1 preliminary safety analysis report was completed. In January 2020, the NNSA approved the construction permit for TMSR-LF1.

The construction of the TMSR-LF1 plant officially began on March 2020, and the plant's main building was completed by the end of 2020. The equipment for TMSR-LF1 has been manufactured, and its installation started by the end of 2020.

5.4. R&D ACTIVITIES IN THE CZECH REPUBLIC

The investigation of nuclear reactor systems with liquid molten salt fuel and corresponding pyrochemical technologies has been carried out in the Czech Republic since 1999. The original impetus for this R&D came from the idea of the Partitioning and Transmutation concept based on the combination of pyrochemical partitioning technology and incineration of transuranic elements in a subcritical Accelerator Driven System with liquid fluoride fuel. After 2005, this idea was gradually abandoned and transformed into the development of the classical MSR technology utilizing the thorium–uranium fuel cycle.

Work in the field of theoretical and experimental development of MSR technology and the thorium–uranium fuel cycle was carried out by a national consortium of institutions and companies managed by ÚJV Řež – Nuclear Research Institute; these R&D activities are led by the Research Centre Řež since 2013.

This R&D has been managed by the Ministry of Industry and Trade of the Czech Republic, which also financially supported two important domestic projects dedicated to MSR technology. These were the projects “Nuclear system SPHINX with molten fluoride salts based liquid nuclear fuel” and “Fluoride reprocessing of GEN IV reactor fuels”, launched in 2004 and 2006, respectively.

Present activities for developing MSR technology are supported by the Technology Agency of the Czech Republic. These activities cover the following fields:

- (a) Reactor physics and neutronics: physics and neutronics of MSR carrier salt, graphite, and comparison of measured data with results from computer codes;
- (b) Fuel and coolant chemistry and supporting technology: liquid fuel processing, pyrochemical separation studies applicable for on-line reprocessing, and studies of flow sheeting;
- (c) Performance of materials: development of alloys based on nickel for the environment of a fluoride salt, mechanical and corrosion tests of these alloys, and development of impellers and flange/gasket systems.

5.4.1. Reactor physics

In the area of theoretical and experimental reactor physics of MSR and Fluoride-salt-cooled High temperature Reactor (FHR), the effort is focused mainly in the interconnection of theoretical and experimental studies of reactor physics of MSRs using the neutron thermal spectrum and neutronic studies of the MSR and FHR. The main part of experimental work concerning the neutronics of a pure FLIBE ($\text{LiF} - \text{BeF}_2$ mixture) salt and the neutronics of FLIBE with thorium and uranium fluorides was carried out at the LR-0 experimental reactor of the Research Centre Řež. The LR-0 core consists of six pin-type fuel assemblies (VVER-1000 design) with nominal uranium enrichment of 3.3% and an empty experimental channel, forming a driven zone in the core centre. Materials are inserted into the driven zone, occupying one position in the lattice. The tests with FLIBE were performed with real MSR $\text{LiF}-\text{BeF}_2$ (66-33 mol %) coolant salt containing the Li-7 isotope (99.994 mol %), which was provided by ORNL, and were aimed at studies of neutron spectrum shape to confirm previous results with $\text{LiF}-\text{NaF}$ salt.

The neutron spectrum behind the layer of salt was measured by the method of recoiled protons in different energy ranges. The independent measurements of neutron spectra were done by a set of hydrogen proportional detectors (HPD) for energies 0.1–1.3 MeV, and by organic scintillators (Stilbene, 10 x 10 mm) for energy ranges 0.8–10 MeV with neutron and gamma pulse shape discrimination. The LR-0 model was analysed with the MCNP6.1 code using data from various nuclear libraries to calculate criticality and neutron spectra. Data from the ENDF/B-VII.0 and VII.1, JEFF-3.1 and 3.2, CENDL-3.1, JENDL-3.3, JENDL-4 and RUSFOND-2010 libraries were then compared with data from older versions of the ENDF/B-VII.0 and JEFF-3.1 libraries. Data from various nuclear libraries were predominantly used to define the material inserted into the reactor zone. These were mainly data related to fuel, moderator, and construction materials so that their influence on criticality could be eliminated, as this was not the main subject of research in these studies. The thermal neutron scattering in FLIBE, TEFLON, and stainless steel canisters was described by the free gas model, and the photo-neutron production was not considered in this physical model [175, 176].

Existing measurements with FLIBE salt were done at room temperature; however, the future neutronic tests will be performed in a special preheated zone put into LR-0. The measurement is planned to be carried out in the temperature range of 500 – 750 °C.

5.4.2. Fuel and coolant chemistry and supporting technology

In the area of chemistry and chemical technology of MSRs, the work covered the verification of processing of liquid fuel, which included the following areas: a) preparation of pure uranium tetrafluoride (UF_4) and thorium tetrafluoride (ThF_4), and b) final processing of MSR liquid fuel samples, that is, preparing several mixture types containing lithium and beryllium fluoride melts (FLIBE) with various contents of uranium and thorium tetrafluorides. The processing of MSR liquid fuel was verified in high laboratory level, which typically corresponds to several hundreds of grams per batch) [177].

The other area of chemical technology covered the study of electrochemical separation from fluoride molten salt. The effort focused on developing an experimental set-up for molten fluoride salt, including developing a reference electrode, and evaluating the individual redox potentials for uranium, thorium, and the most important fission products in selected molten

fluoride salts – typically LiF-BeF₂ (FLIBE), LiF-CaF₂ and LiF-NaF-KF (FLINAK) mixtures. Another development focused on investigating the technique of fused salt volatilization proposed for the extraction of uranium (in the form of UF₆) [178].

In addition, studies of the technology of the MSR fuel cycle covered the flow-sheeting research focused on balance-simulation calculations and proposals of reprocessing diagrams, safety assessment, and non-proliferation and physical protection aspects of the systems used in the thorium–uranium fuel cycle.

5.4.3. Performance of materials

The main part of research on materials focused on developing the special nickel superalloy MONICR, which is suitable as a structural material for the MSR technology. The mechanical tests, corrosion and irradiation tests have been carried out, and the development is also focused on studying casting, forming and recrystallization. Other studies are focused on the use of MONICR for the manufacture of components of the MSR technology, such as impellers, flanges, and valves [179]. The molten fluoride loop programme, which is focused mainly on dynamic corrosion tests of structural material and verification of the function of selected devices and methods of measurement in the environment of molten salts, is an integral part of the investigation. An important part of this programme has been also the design and study of the operation of the so-called freeze safety valve.

5.5. R&D ACTIVITIES IN DENMARK

Until only a few years ago, research on nuclear technologies in Denmark had been carried out solely at the Technical University of Denmark's (DTU) Centre for Nuclear Technologies. Only recently, commercial Danish designs have been proposed. Research and development in MSR technologies in Denmark is chiefly carried out at two nuclear start-ups, Copenhagen Atomics and Seaborg Technologies, and at DTU.

In part based on the merger of the Risø National Laboratory with DTU in 2007, DTU has a long history of research in nuclear energy. Originally centred around the three research reactors at Risø, in particular the reactor DR3 that was cooled by heavy water and generated 10 MW(th), nuclear research originally focused on reactor physics, nuclear chemistry, dosimetry and health physics, and radioecology. For many years, DR3 constituted an important neutron source in Europe, utilized for a range of applications within materials research, neutron scattering experiments, isotope production, and neutron transmutation doping of silicon for high power semiconductors. After the closure of DR3 in 2000, materials science with neutrons continued at DTU and at the University of Copenhagen using other neutron scattering facilities in Europe and elsewhere.

Meanwhile, research at the Risø National Laboratory took a broader perspective, and as Denmark shifted away from including nuclear power in its energy policy, R&D expanded to cover a range of renewable energy technologies including wind, solar PV cells, fuel cells and electrolysis, and bioenergy; nuclear technologies research focused on radioecology, dosimetry, luminescence physics and neutronics.

Neutronics at DTU today encompasses three main areas: i) physics of neutron scattering, in particular aimed at code development (e.g. McStas, MCPL, NCrystal), neutron scattering

kernels, and neutronics support for neutron scattering experiments, ii) neutronics design, aimed at designing and optimizing nuclear scattering instruments, neutron optics, moderators, and nuclear diagnostics, and iii) reactor physics. Activities within all these fields are carried out in close collaboration with external partners, in particular with the European Spallation Source (target monolith design including the moderator-reflector system and target wheel diagnostics), Fusion for Energy (neutronics design of the Collective Thomson Scattering diagnostic for ITER), and Seaborg Technologies (MSR development projects).

Research on reactor physics (the third area) focuses on model development for MSR design. In its main design version, the MSR has the fuel dissolved in the molten salt, resulting in strongly coupled neutronic and thermohydraulic effects, further complicated by the transport of delayed neutron precursors by the circulating fuel. In an ongoing PhD project, different coupling schemes of neutronic-thermohydraulic single physics codes are investigated, using open source software (OpenFOAM CFD) and restricted neutronics codes (Serpent, MCNP). The different coupling approaches are benchmarked against MSRE historical data as well as with data from molten salt experimental loops, such as the DYNASTY loop at Politecnico di Milano, Italy.

In the SEALION collaborative project (with Seaborg Technologies, the Chalmers University of Technology in Sweden, and a Swiss start-up, DAES), regulatory compliant multiphysics software is being developed intended to ease the process of licensing MSR designs. The approach taken is to apply an external coupling scheme between the neutronic and thermohydraulic codes, leaving these codes of single physics unaltered. The coupled codes are benchmarked against MSRE.

A new reactor technology project has been started in 2020 at DTU, also in collaboration with Seaborg Technologies, to develop designs of freeze valves for MSRs. The project aims at developing and applying CFD models for optimizing and carrying out safety analysis of these designs.

Copenhagen Atomics is a Danish molten salt technology and reactor company having the aim of developing thorium breeder MSRs that are mass manufacturable. The company was formed in 2014 and incorporated in 2015 and was the first nuclear reactor company in Denmark. Contrary to most other reactor development companies is that Copenhagen Atomics chose a unique approach to developing a reactor where all components of the design are developed, assembled, and tested inhouse, and the systems as a whole tested with non-nuclear salts, thus allowing for rapid iterations and reliable systems before going through the nuclear regulatory approval of a design that has never been built and tested. Copenhagen Atomics promotes open collaborations with other companies, universities, and national laboratories and offers many of its test systems for purchase and for academic research. The objective is to accelerate the development of molten salt technologies and help talented people get hands-on experience with molten salt systems.

Copenhagen Atomics is actively developing components considered to be state of the art for MSRs, including a canned rotor molten salt pump with electromagnetic bearings, molten salt valves, molten salt filters, molten salt flow meters, qualification of commercial heat exchangers for use in MSRs, salt spraying off-gas systems, qualification of composite materials as salt wetted structural material in MSRs, fluoride salt purification, online molten salt redux meter, online laser breakdown spectroscopy meter, autonomously reactor control systems and

software, assembly line production of molten salt test loops, and MSR that are container sized. Part of these research activities are carried out with university and industry partners.

Copenhagen Atomics also received approval by a national nuclear regulatory body to work with a salt containing fertile uranium and thorium at a facility outside of Denmark, and is planning to run static salt corrosion experiments and pumped salt loop tests of long duration with fertile bearing salts starting in 2021.

Copenhagen Atomics is developing a 100 MW(th) breeder MSR that is moderated with heavy water (thermal spectrum) and that fits inside a leak-tight containment, which is sized the same as a shipping container with a length of 40 feet. The heavy water moderator is unpressurized, thermally insulated from the fuel salt that can reach up to 700 °C, continually circulated and cooled, and passively drained from the core in case a loss of power or abnormal condition is detected. The requirement for materials that are highly reliable with low absorption of neutrons for separating and insulating the fuel salt and moderator is an ongoing research activity at Copenhagen Atomics that is being carried out in collaboration with university and industry partners.

In 2014, Seaborg Technologies proposed the Molten Salt Thermal Wasteburner (MSTW), a thermal spectrum, graphite moderated, MSR with a single salt, fuelled with a combination of spent nuclear fuel (SNF) and thorium [180]. It was envisioned to produce 115 MW(e) with a two-stage turbine, from 270 MW(th). The design philosophy was to create a reactor that was small, modular, and that could be produced in large quantities, with a lifetime of about seven years. The design developed into the current Compact Molten Salt Reactor (CMSR), a modular reactor generating 250 MW(th), placed on a floating barge, and whose specifications are discussed in Subsection V.2.8. The conceptual design phase of the CMSR was completed with the first regulatory approval phase finalised as well in 2020. The current focus of the company is on molten salt experiments, building and testing equipment prototypes, modelling and simulation, and developing the basic engineering design.

5.6. R&D ACTIVITIES BY THE EUROPEAN COMMISSION

5.6.1. Introduction and main achievements

5.6.1.1. *History*

The research efforts of The Joint Research Centre (JRC) of the European Commission on the technology of MSRs started around 2004. At that time, a group of research organisations in Europe started a collaboration on this type of a nuclear reactor, which was selected by the Generation IV Forum. This collaboration resulted in the MOST project in the 5th Euratom Framework Programme. The JRC activity started in Karlsruhe, a research installation with a strong focus on fuel and fuel cycle issues. Naturally, JRC thus started to address the chemistry of the molten salt fuel.

The liquid MSR fuel is a liquid solution in which chemistry plays a much more important role and thermomechanics a minor role, unlike solid oxide fuel. The temperature distribution in the liquid is relatively homogeneous, and the mutual solubility in the liquid of the salt and the fission products is high. As a result, the chemistry and some of the properties of the liquid MSR

fuel can be described very well by equilibrium thermodynamics. This has been the prime focus of the JRC research in the first period (2002-2006).

From the beginning, a combination of thermodynamic modelling and experimental studies has been the goal. The extensive literature on phase diagrams and thermodynamic properties of molten salt systems from the MSRE and MSBR period at ORNL gave a solid base to develop an extensive thermodynamic model of MSR fuel systems with the CALPHAD (CALculation of Phase Diagrams) approach. In this approach, the experimental data are fitted to semi-empirical models for the various phases (solids, liquids, gases) to obtain a consistent description of multicomponent systems. Such a model, if based on sufficient and reliable experimental data, has a strong predictive capacity, and can be used for design and safety analysis. The focus was on the $\text{LiF-BeF}_2\text{-ThF}_4\text{-UF}_4$ system.

Initially the experimental work was aimed at validation of data from ORNL and other sources, and was extended when needed to improve the reliability of the models. However, the experimental work quickly proved to be complicated. Fluoride salts are corrosive, reactive and although only slightly volatile, experimental equipment needs appropriate protection. Moreover, methods needed to be developed to work with small quantities (not the gram-quantities used by ORNL), and quickly became clear that the samples of ThF_4 and UF_4 that were made available to JRC were not sufficiently pure because they contained impurities that affected the measurements. Thus, the experimental work originally focussed on two areas: the synthesis of pure materials (Subsection 5.6.2.1), and the development of the appropriate experimental tools for accurately measuring physical properties at high temperatures (Subsection 5.6.2.4).

As a result, JRC at Karlsruhe (JRC-Karlsruhe), Germany, now has a suite of instruments and facilities that are suited for investigating molten salt fuels, as described in this subsection. These facilities have been used extensively in a series of the Euratom indirect action funded projects, as described in Subsection 5.6.1.2. In addition, a comprehensive thermodynamic database has been compiled, the JRCMSRD, which is nowadays available through the TAF-ID project of the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD). The experimental and modelling studies are well embedded in collaborations with European academic, research and industrial organisations, but also outside of the European Union (EU), and address a wide range of different fuels and coolants, including fluoride and chloride salts. The ongoing work has evolved/extended to irradiation experiments in the High Flux Reactor (HFR) (see Subsection 5.6.3), in which molten salts are irradiated in a reactor for the first time since several decades ago. These experiments, which are described in Subsections 5.6.2.4 and 5.10.1.6, will provide further samples to test the thermochemical models and will provide insight into the fuel behaviour and how the fuel chemistry model will need to be developed further.

5.6.1.2. European Framework projects

In the frame of the indirect action of the Euratom Research and Training programme, a series of projects on MSR technology have been supported since 2004, as shown in Figure 18. These projects showed an evolution from research on the ORNL concepts (MOST), to a broader review of molten salt applications in nuclear technology, to a focus on the Molten Salt Fast Reactor (MSFR) design by the CNRS. Research within these projects is being carried out in various countries of the European Union (EU) in the framework of national research programmes combined with European Framework R&D programmes.

FIG. 18 Time evolution of EU-projects on MSR R&D.

After completion of the MOST project, the ALISIA (Assessment of LIquid Salts for Innovative Applications) project was funded for one year and was dedicated to exploring the potential of MSR in the nuclear field. Furthermore, it aimed at the selection of the fuel salt and design choices for a European MSR.

The EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor System) project focussed on the integral evaluation of the MSFR design (which at that time utilized both core fuel and blanket salt that were rich in thorium to breed fissile uranium), including reactor and clean-up facility, and led to a build-up of necessary computational tools in Europe. It was performed in close collaboration with the sister Russian project MARS, which was sponsored by ROSATOM, in the frame of the EU-Russia collaboration in the field of nuclear energy. The conclusions of EVOL can be summarised as follows:¹⁴

- An innovative design was proposed based on safety analyses combined with thermohydraulic studies and neutronic calculations. Previous studies highlighted the necessity to actively circulate and cool the blanket salt. In the present design option, the blanket salt flows in a circuit external to the salt container. This arrangement ensures a very good neutron economy because most of the neutrons escaping from the reactor core will be captured in the surrounding blanket. Moreover, the blanket salt follows a path that improves the cooling of the main metallic structures of the fuel circuit;
- Since ^{233}U is not available, the start-up of the MSFR needs another fissile material, that is, ^{235}U and/or ^{239}Pu (or transuranic elements). Two options for the fuel composition have been proposed that meet the requirements from the points of view of neutronics, melting temperature, physico-chemical properties (such as viscosity, density, and solubility of elements), redox potential (required for material compatibility), fuel reprocessing, and economy. The two options retained are: $\text{LiF-ThF}_4\text{-UF}_4\text{-PuF}_3$ (78.6-12.9-3.5-5 mol%) and $\text{LiF-ThF}_4\text{-UF}_4\text{-(TRU)F}_3$ (77.5-6.6-12.3-3.6). The two options require an enrichment of ^{238}U into ^{235}U respectively equal to 20 and 13%;
- The fuel reprocessing scheme has been achieved and the chemical database required for calculations of efficiency of the involved processes was established in the frame of the project. New steps have been proposed for the reprocessing, leading to efficiently recovering the fuel components. The behaviour of all relevant fission products in the reprocessing has been evaluated.
- Three heats of NiWCr alloys were manufactured in the project. Based on studying them, the optimized composition for MSFR is (in wt%): W 21.7 - Cr 7.5 - Mn < 0.02 - Al 0.22 - Zr 350 (ppm) - Ti 0.22. The corrosion studies have evidenced the role of oxygen and of the ratio U(IV)/U(III) to decrease the corrosion reactions. Dynamic studies have been performed through ROSATOM/EURATOM collaboration.

¹⁴ European Commission, Evaluation and Viability of Liquid Fuel Fast Reactor System, <https://cordis.europa.eu/project/id/249696>

In the SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor) project, the key safety features of the MSFR were analysed in depth, based on substantial improvement of computational tools, particularly in the field of multi-physics analysis, and experimental studies. The conclusions of SAMOFAR can be summarised as follows:

- Initiating events have been identified and classified;
- Properties of the MSFR fuel have been well defined;
- Valuable experimental tools for validating simulation tools have been implemented;
- Enormous progress has been made in the field of multi-physics modelling of the core;
- The potential of the application of safety barriers have been further substantiated.

The latest project of the European Community is the SAMOSAFER proposal, which is a logical follow up of the previous R&D projects funded by the EU. The SAMOSAFER project started in 2019 (project of 4 years ending in 2023), and it will build upon the experience and knowledge from the previous projects with the aim to ensure that MSR technology (not only limited to the MSFR concept studied in the EU in the past) can fully comply with the more stringent safety requirements expected in the next 30 years. In particular, SAMOSAFER will focus on the following phenomena important for safety assessments:

- Freezing of the fuel salt against cold walls and subsequent re-melting of the salt;
- Internal heating of the fuel salt causing lower natural circulation and local overheating;
- Overheating of the fuel salt in the MSFR core during transients and in the drain tanks during storage;
- Effects of transients on the thermo-mechanical integrity of the primary circuit;
- Redistribution of the inventories of radionuclides in the Fuel Treatment Unit (FTU) via gas bubbling, fluorination, and chemical extraction leading to changes in chemistry and mobility of radionuclides.

The chemistry and physics needed to increase the safety of the MSFR, and of a MSR in general, will also be modelled in SAMOSAFER, such as:

- Thermo-chemical modelling to evaluate the properties of the fuel salt to retain fission products, and the effects of fission products and corrosion products on the fuel salt's thermo-physical properties (such as melting point, heat capacity, vapour pressure, viscosity, and thermal conductivity);
- Radiation heat transfer to accurately calculate the decay heat removal from the Emergency Drain System (EDS);
- Predictive reactor control strategies to reduce the number of draining events, to avoid accidents, and to mitigate their consequences;
- Redox control of the fuel salt to reduce corrosion in the primary circuit;
- Reactor scaling effects on the safety of nuclear reactors in general;
- Uncertainty quantification methods based on non-intrusive polynomial chaos expansion and reduced order methods (important for licensing).

New safety features to prevent and mitigate severe accidents also will be designed, such as:

- Iodine trapping system, off-gas system, and the gas holdup system of the gaseous FPs in the FTU;
- Freeze plugs and drain tanks (drain tanks for both emergency and normal conditions).

For the experimental validation of the new simulation methods and tools, the following facilities will be used:

- DYNASTY: This is a facility at POLIMI with a height of 10 meters to study the flow dynamics of internally heated salts. Two versions exist: a single loop system and two connected loops. The latter setup simulates the primary salt circuit (core region) connected to the salt flow in the intermediate circuit. DYNASTY is fully instrumented and can be used to study decay heat removal from the core region by natural circulation. DYNASTY has been modified in the SAMOFAR project to fully meet the specific needs of MSRs;
- ESPRESSO: This is a new facility at TU Delft to investigate melting and solidification in conjunction with flow. It consists of a transparent container with cooling options at the bottom. Both the container and the fluid are transparent, which enable optical access to determine local velocities in the salt with laser methods like Laser Doppler Anemometry (LDA) and Particle Image Velocimetry (PIV), and the development of the solid layer near the cooled surface;
- High Flux Reactor: This is a material testing reactor in Petten capable of generating 40 MW(th) and being used to irradiate samples of fuel salt consisting of ThF₄ mixed in LiF. After the irradiation of 1-2 years, the samples will be analysed both at NRG and at JRC-Karlsruhe, focusing on the composition of fission products in the samples, Redox potential, and the interactions between the fuel salt and the graphite crucibles and between the fuel salt and the metal encapsulation;
- SWATH-S: This is a facility at CNRS consisting of two vessels of which one is filled with liquid salt (FLiNaK). The salt can flow by pressure from one vessel to the other, thereby passing through an experimental station in a glovebox. The experimental station can be adapted to the needs of the experiments, like the measurement of flow and temperature profiles of the salt in turbulent and laminar conditions, and freezing phenomena of the salt against cold walls.
- Experimental facilities of JRC (described in detail in Subsection 5.6.2.4) will be used to provide novel data on physico-chemical properties of MSR fuels.

Simulation tools used in SAMOFAR and SAMOSAFER. For the assessment of the MSFR safety during various transient scenarios, multi-physics tools are necessary to capture the appropriate physics of the reactor system. Classical codes used in reactor physics cannot be used as they do not allow for the key features of the MSFR. The special features of the MSFR include the movement of precursors with the moving fuel, the strong coupling between the neutronic and the thermohydraulic aspects due to the use of liquid fuel, the internal heat generation and the shape of the core having no fuel pins as a repeated structure. These features cause a variety of phenomena occurring during transients that are particular to the MSFR for which dedicated tools have been developed. This subsection presents a basic description of the tools used in the SAMOFAR and SAMOSAFER projects. The capabilities of these codes include detailed computational fluid dynamics analysis (2D, 3D, geometrically flexible) detailed neutronics (diffusion, transport), and incorporation of complex physics such as gas bubbling, and the fuel salt's melting and solidification. Specifically, the code systems of CNRS, KIT/EdF, TU Delft, Politecnico di Milano (PoliMi), and PSI are described.

KIT (SIMMER-III). SIMMER-III is a two-dimensional, multi-velocity-field, multiphase, multi-component, Eulerian fluid-dynamics code coupled with a fuel-pin model and a space- and energy-dependent neutron kinetics model. In this code, phase change (melting/freezing; evaporation/condensation) models are available. This code consists of three basic elements: 1)

fluid-dynamics model, 2) structure (fuel-pin) model, and 3) neutronics model. The fluid-dynamics model (the main part of the whole code) is interfaced with the structure model through heat and mass transfer at structure surfaces. The neutronics model provides nuclear heat sources based on the time-dependent neutron flux distribution consistent with the mass and energy distributions. The SIMMER-III code models incorporate five basic core materials: fuel, steel, coolant, control and fission gas. A material can exist in different physical states; for example, fuel can be represented as fabricated pin fuel, liquid fuel, a crust refrozen on structure, solid particles, broken fuel pellets (called chunks) and fuel vapor, whereas fission gas exists only in a single gaseous state. The fuel includes fertile and fissile components to represent different enrichment zones in the core. Material density and energy component distributions are obtained through mass, momentum and energy conservation equations. Multi-velocity-field formulation and the fluid convection solution algorithm are based on a time-factorization approach.

PoliMi (OpenFoam). A multiphysics solver for the analysis of the accidental scenarios of the MSFR has been developed and extended, starting from a previous tool based on the OpenFOAM library [181]. OpenFOAM is an open source library for numerical simulation in continuum mechanics using the Finite Volume Method. The toolkit is very flexible thanks to the object-oriented programming, allowing users to customize, extend and implement a complex physical model [182]. OpenFOAM facilitates the parallelization of the developed solvers thanks to dedicated routines based on geometrical domain decomposition.

The solver allows for the transient analysis of the MSFR, describing the neutronics and the thermohydraulic behaviour of the reactor, the movement of the delayed neutron precursors, and the decay heat. With respect to the previous version, the solver has been extended by introducing new features for neutronics and thermal hydraulics. Regarding the neutronics, a multi-group diffusive model has been adopted along with some acceleration routines in order to speed up the convergence rate of the multi-group neutronics solution. This improvement was achieved by implementing in the PoliMi solver the neutronics module of the Gen-FOAM multiphysics platform [183, 184], which is a general-purpose solver for transient analysis of nuclear reactors, thanks to the object-oriented programming of OpenFOAM. On the thermohydraulic side, the code was extended by introducing the compressibility, which could have an important effect in the simulation of fast transients [185]. In addition, the transport of gas bubbles in the salt mixture was added to simulate the effect of increasing/decreasing bubbling injection on the reactor dynamics. Indeed, besides the removal of fission products, the bubbling system can be adopted for the MSFR reactivity control.

PSI (GeN-Foam). GeN-Foam is a reactor modelling code based on OpenFOAM, and is a library written in C++ used to create executables that solve problems of continuum mechanics or perform data manipulation. It is a multiphysics code capable of modelling neutronic, thermohydraulic, and thermomechanical aspects in steady state and transient conditions. Features of GeN-Foam have already been described in detail in journal publications from the main code developer [183, 184].

TU Delft (PHANTOM-SN and DG-Flow). The tools in the area of neutronics have evolved into full transport codes based on discontinuous Galerkin finite elements (DGFEM) at TU Delft. The in-house code based on this principle is PHANTOM-SN and can be considered state of the art solver technology. A similar CFD code based on DGFEM has been developed and is called DG-flow, that combines the advantage of local conservation as in finite volumes with the geometric flexibility and high-order accuracy of finite elements. The coupling of these two

tools is capable of high accuracy simulation of transients in the MSFR including the movement of the precursors. In the future, additional physics will be included such as the tracking of bubbles and incorporation of the bubbles effect on the fluid phase.

PHANTOM-SN. It is a radiation transport tool developed at TU Delft capable of solving the Boltzmann radiation transport equation (Linear Boltzmann equation (LBE)). The code has extensive capabilities, such as solving the steady state and time-dependent LBE, both principal and multimodal calculations of various eigenvalue types (criticality and time-eigenvalues), and extensive perturbation analysis (both regular and generalized).

DG-flow is a Navier-Stokes solver based on the discontinuous Galerkin approach and developed at TU Delft. Both standard Navier-Stokes as well as turbulent flow can be handled by using Reynolds-Averaged Navier Stokes models (RANS).

CNRS (Transient Fission Matrix–OpenFOAM). The Transient Fission Matrix (TFM) approach [186, 187], described in Subsection 5.7.2.2, has been developed specifically as a neutronic model to account for the phenomena associated with precursor motion and to perform coupled transient calculations close to the Monte Carlo accuracy for the neutronics while sustaining a low computational cost. This approach is based on a pre-calculation of the neutronic reactor response through time prior to the transient calculation. The results of the calculations using the SERPENT Monte Carlo code [188] are condensed in fission matrices, keeping the time information. These matrices are interpolated to account for the effects of local Doppler and density thermal feedbacks due to temperature variation in the reactor system. This approach provides very quickly an estimation of the neutron flux variation for any temperature and precursor distribution in the reactor.

5.6.2. Fuel and coolant chemistry and supporting technology

5.6.2.1. Synthesis of actinide halides

Basic thermodynamic and electrochemical data of actinide fluorides and chlorides are required for the design and safety assessment of any MSR concept. Since these compounds are usually not available commercially, or not in the required purity, they have to be prepared from the available input materials, typically oxides. This subsection summarises the activities on synthesis and characterisation of actinide fluorides and chlorides done in JRC-Karlsruhe.

Synthesis of actinide fluorides. Equipment for synthesis of pure actinide fluorides has been designed and installed in JRC-Karlsruhe, and it uses pure hydrogen fluoride (HF) gas as the fluorination agent. It consists of a glove box kept under purified inert Ar gas atmosphere with oxygen and moisture content < 1 ppm, a fluorination reactor at high temperature, and a HF supply gas line connected to the glove box. The fluorination apparatus is a horizontal tube reactor made of Inconel X-750 inserted in a resistance furnace enabling work at temperatures up to 1200°C. A scheme and a photo of the fluorination reactor is in Figure 19 and the equipment is in detail described in [189].

FIG. 19 A scheme and a photo of the fluorination reactor installed in JRC Karlsruhe.

The syntheses are based on a solid-gas reaction between the stoichiometric oxides with pure HF gas at elevated temperatures. The oxide powder in batches up to 15g is inserted into a boron nitride boat and placed in the fluorination reactor, which is heated to the desired temperature. HF gas is then led directly above the boat in a molar excess 3-4 providing the conversion of oxide to fluoride. The non-reacted HF gas and the possibly formed gaseous products are removed to the off-gas treatment unit by a stream of pure argon carrier gas, introduced to the reactor through a port in a front flange. If needed, the synthesis can be completed by consequent reduction of the fluorination product by a mixture of 6% H₂ in dry argon gas at the desired temperature. A summary of the conditions for the established syntheses is given in Table 11, photos of the final products are shown in Figure 20 and detailed descriptions are in Refs [189, 190, 191].

TABLE 11. THE SYNTHESSES OF ACTINIDES FLUORIDES ESTABLISHED IN JRC KARLSRUHE

Product	Chemical reaction	Temperature
ThF ₄	ThO ₂ (s) + 4HF(g) → ThF ₄ (s) + 2H ₂ O(g)	600°C
UF ₄	UO ₂ (s) + 4HF(g) → UF ₄ (s) + 2H ₂ O(g)	450°C
UF ₃	2UF ₄ (s) + H ₂ (g) → 2UF ₃ (s) + 2HF(g)	800°C
PuF ₃	PuO ₂ (s) + 4HF(g) → PuF ₄ (s)* + 2H ₂ O(g)	550°C
	2PuF ₄ (s) + H ₂ (g) → 2PuF ₃ (s) + 2HF(g)	600°C

* PuF₄ cannot be obtained as a pure product, but as a mixture of PuF₄ and PuF₃ due to observed partial reduction during the fluorination process, likely due to traces of hydrogen in the HF gas that was used.

FIG. 20. Appearance of the pure actinides fluorides synthesised in JRC Karlsruhe.

A combination of the following techniques is used for the quality control of the synthesised fluorides: gravimetric mass balance of the reaction, X-ray diffraction (XRD) for the qualitative phase analysis, differential scanning calorimetry (DSC) for the melting temperature determination, and inductively coupled plasma mass spectrometry (ICP-MS) for the metal bases impurities. ThF₄, UF₄ and PuF₃ were synthesised in JRC Karlsruhe with phase purity 100 ± 0.5% (uncertainty of the methods) and with metal base purity > 99.8% [189], [190]. UF₃ had phase purity 99.6% ± 0.5% based on XRD, while DSC and ICP-MS analyses were not available at the date this report was published.

The irradiation experiments SALIENT-01 and -03, which are described in Subsections 5.10.1.6 and 5.6.3, respectively, are being carried out in collaboration between the Nuclear Research and Consultancy Group (NRG, the Netherlands) and JRC Karlsruhe. The actinide fluoride mixtures required for these experiments were synthesised in JRC Karlsruhe by the methods described above, as summarised in Table 12.

TABLE 12. MATERIALS SYNTHESISED IN JRC KARLSRUHE FOR SALIENT IRRADIATION EXPERIMENTS

Salt mixture	Composition (mol. %)						Mass (g)	Experiment
	⁷ LiF	LiF	ThF ₄	UF ₄	UF ₃	PuF ₃		
⁷ LiF-ThF ₄	78.0	-	22.0	-	-	-	28.6	SAL-1
⁷ LiF-ThF ₄ -UF ₄ -PuF ₃	75.0	-	18.7	6.0	-	0.3	28.0	SAL-2
⁷ LiF-ThF ₄ -UF ₄ -UF ₃ -PuF ₃	75.0	-	18.7	5.7	0.3	0.3	15.0	SAL-2
⁷ LiF-ThF ₄ -UF ₄ -CrF ₃ -PuF ₃	74.7	-	18.6	6.0	-	0.3	7.0	SAL-2
LiF-ThF ₄ -UF ₄ -PuF ₃	-	74.9	23.0	2.0	0.1	-	50.0	SAL-2

Synthesis of actinide chlorides. Equipment for synthesis and purification of actinide chlorides was designed and installed at JRC-Karlsruhe in 2019 and it consists of two connected glove boxes, a chlorination reactor at high temperature, Cl₂/HCl gas line connected to the main glove box and an off-gas treatment unit in the auxiliary glove box. The chlorination reactor is a gas-tight quartz tube closed by Inconel flanges inserted in a horizontal resistance furnace enabling work at temperatures up to 800°C. A scheme and a photo of the reactor are shown in Figure 21. The reactor is intended for a solid-gas reaction, the starting powdered material is inserted in a quartz boat to the reactor, and the powder stays in the boat during the complete reaction. The set-up enables flow-through chlorination, while Cl₂ or HCl gas is led directly above the boat and the second reactive gas (e.g., CCl₄ vapours in Ar, H₂(6%)/Ar mixed gas or an inert carrier gas) are introduced into the reactor through the flange inlet. The excess reaction gases and the possibly formed gaseous products are removed to the off-gas unit. The equipment is in detail described in [192].

Procedures for syntheses of actinide chlorides are being investigated and optimised. At the time this report was published, a novel method for preparation of UCl₄ from stoichiometric UO₂ using combination of Cl₂ and CCl₄ gases was proposed and tested, while syntheses of other actinide chlorides are in the preparatory phase. As reported and confirmed by thermodynamic calculations, UCl₄ cannot be formed by a simple reaction of UO₂ with Cl₂ or HCl gases at temperatures up to 1000°C, as UOCl₂ is always predominantly formed. In addition, the reaction of UO₂ with CCl₄ gas is efficient only at elevated temperatures, when higher chlorides (i.e. with higher valency) are formed together with UCl₄. The proposed method is thus based on combining Cl₂ and CCl₄ gases as chlorination agent and a source of carbon to convert UO₂ to UCl₄ at temperatures lower than needed for the reactions with pure CCl₄.

FIG. 21. A scheme and a photo of the chlorination reactor installed in JRC Karlsruhe.

A phase pure UCl₄ product was obtained using two-step chlorination at a temperature of 175°C, followed by purification of the intermediate chlorination product composed of a UCl₄/UO₂+x mixture by sublimation at 600°C, and recovery of the final UCl₄ product at a cold-finger at 400°C. The quality control of the obtained products is based on a combination of the same

methods as described above for the synthesis of actinide fluorides. Photos of the initial material, the chlorination and final products, and the XRD pattern of the recovered phase-pure UCl_4 are shown on Figure 22, and a detailed description of the method and the results are published in Ref. [192].

FIG. 22. Photos of the initial material, the chlorination and final products, and an XRD pattern of the phase-pure UCl_4 final product.

5.6.2.2. Purity control of the samples

After every synthesis it is essential to check for the purity of the samples. Two types of purities are generally considered, phase purity and chemical purity. It has been demonstrated in the past that four independent methods are used to verify the sample's purity. The first one is the mass balance check knowing the chemical state (and purity) of the initial material and the chemical state of the product. This controlling step is accompanied with visual check of the formed product (such as colour, homogeneity of colour, and powder form).

The second purity check is done through XRD analysis for phase purity including a phase quantification by Rietveld refinement. Figure 23 shows two X-ray diffractograms of synthesised UF_4 : (i) one containing residual UO_2 (as initial material before fluorination), and (ii) one identified as UF_4 that is phase pure. Obviously the latter one doesn't contain evidence of the UO_2 -related peak.

It was however observed in the past that even samples that were considered as XRD pure (thus no evidence of other phase impurities) still contained some impurities, such as oxy-fluorides or oxides, that could significantly influence their behaviour. Therefore, a third purity check is being performed for each compound, a melting point determination done by Differential Scanning Calorimetry (DSC). This method has been demonstrated as a very sensitive technique for impurities detection. Figure 24 gives an example showing two melting point determinations of purified and not-purified ThF_4 . While only one peak is observed on the heat flow signal of DSC with correct melting point (not quantified in the figure) for the pure ThF_4 compound, the one containing traces of impurities revealed two peaks and lower melting point, clearly indicating presence of second phase.

The fourth analysis is performed by ICP-MS and is in place to quantify concentrations of chemical elements.

All four methods are part of the protocol for quality assurance used at JRC for safety studies and can be presented to regulators.

FIG. 23. XRD analysis of purified and not-purified UF_4 .

FIG. 24. Melting point determination of purified and not-purified ThF₄ determined by DSC.

5.6.2.3. *Electrochemistry of molten salts*

Electrochemical characterization of chlorides. A systematic experimental study was carried out in JRC-Karlsruhe to obtain accurate knowledge of the thermodynamic and electrochemical properties of actinides (An) dissolved in a molten LiCl-KCl eutectic salt [193, 194, 195, 196, 197, 198]. Apart from data on the general electrochemical behaviour (such as electrode potentials and reaction mechanisms), thermo-chemical data (activity coefficients, enthalpies and entropies of formation) and kinetic data (diffusion coefficients) were evaluated for all actinides up to Am. Their electrochemical properties were measured on different inert and reactive working electrode materials.

A dedicated equipment for electrochemical measurements in molten chloride media was designed and installed in JRC Karlsruhe. It consists of a glove box under purified argon atmosphere, equipped with a vertical well-type oven with an electrolyser made of stainless steel inside. The electrochemical cell is an alumina or quartz crucible containing the measured melt and three electrodes, positioned through a water-cooled top flange of the electrolyser. A working electrode is typically a metallic wire with diameter 1 mm, a counter electrode is made from a Mo wire bent into the shape of a spiral, and a reference electrode is a silver wire dipped into a LiCl-KCl-AgCl (1 wt.%) melt in a Pyrex glass tube. The glove box, a scheme, and a photo of the set-up is shown in Figure 25. The equipment is described in detail in Refs [194, 196].

Transient electrochemical techniques, such as cyclic voltammetry, chronopotentiometry, square-wave voltammetry and similar were used to measure the above listed properties. The selected properties of actinides in molten eutectic LiCl-KCl measured on inert working electrodes are summarised in Table 13 as compiled from Refs [193, 194, 195, 196, 197, 198, 199, 200]. In addition, electrochemical formation of actinide alloys with reactive materials of different working electrodes was investigated using mainly solid aluminium and liquid cadmium electrodes. A comparison of measured apparent standard potentials of selected actinides and lanthanides on inert, solid reactive and liquid reactive electrodes is illustrated in Figure 26 [199].

FIG. 25. Glove box for electrochemistry in molten chlorides installed in JRC Karlsruhe (left) and a scheme and a photo of the electrochemical cell (right).

TABLE 13. SELECTED PROPERTIES OF ACTINIDES IN MOLTEN LiCl-KCl EUTECTIC AT T = 450°C EXPERIMENTALLY DETERMINED IN JRC KARLSRUHE (COMPILED FROM REFS [193, 194, 195, 196, 197, 198, 199, 200])

Actinide	Th	U		Np		Pu	Am	
Reduction mechanism	1 step	2 step		2 step		1 step	2 step	
Redox systems	Th ⁴⁺ /Th ⁰	U ⁴⁺ /U ³⁺	U ³⁺ /U ⁰	Np ⁴⁺ /Np ³⁺	Np ³⁺ /Np ⁰	Pu ³⁺ /Pu ⁰	Am ³⁺ /Am ²⁺	Am ²⁺ /Am ⁰
Exchanged electrons, <i>n</i>	4	1	3	1	3	3	1	2
App. standard potential, <i>E</i> ^{0*} (V vs. Ag/AgCl 1 wt.%)	-1.36	-0.23	-1.28	+0.46	-1.50	-1.57	-1.47	-1.67
Gibbs free energy form., <i>ΔG</i> ^{∞_{MClx}} (kJ/mol)	-987.6	-137.8	-726.2	-69.87	-776.4	-801.9	-816.6	-558.3
Specie	Th ⁴⁺	U ⁴⁺	U ³⁺	Np ⁴⁺	Np ³⁺	Pu ³⁺	Am ³⁺	Am ²⁺
Activity coefficient, <i>γ</i> ⁰	4.6×10 ⁻⁴	1.5×10 ⁻²	5.0×10 ⁻⁴	8.5×10 ⁻³	3.0×10 ⁻⁵	~1.0×10 ⁻²	4.7×10 ⁻³	2.7×10 ⁻²
Diffusion coefficient, <i>D</i> (cm ² /s)	3.2×10 ⁻⁵	2.1×10 ⁻⁵	2.7×10 ⁻⁵	2.1×10 ⁻⁵	1.9×10 ⁻⁵	1.6×10 ⁻⁵	2.4×10 ⁻⁵	1.2×10 ⁻⁵

Results of these electrochemical studies supported the design of an electrorefining process for group-selective recovery of actinides from nuclear fuel using reactive solid aluminium cathodes [201, 202]. In this process, electroseparation of actinides from fission products is carried out in LiCl-KCl eutectic molten salt at a temperature of 450°C by applying a constant current between a metallic fuel contained in an anodic basket and an aluminium cathode. During the process, actinide cations arising from the anodic oxidation of the fuel are electro-transported and deposited onto the aluminium cathode where they form An-Al alloys, while fission products either stay in the melt or they are not dissolved from the fuel. The process was intensively studied in JRC Karlsruhe and successfully demonstrated in a laboratory scale using irradiated experimental metallic fuel METAPHIX [203, 204]. An excellent performance of the solid aluminium cathode was shown, as very high selectivity of actinides over lanthanides was reached, as well as sufficient efficiency of the process and high capacity of aluminium to take up actinides.

FIG. 26. Apparent electrochemical potentials measured in JRC Karlsruhe for various actinides and lanthanides using different working electrodes in LiCl-KCl eutectic at 450°C. The represented values are based on the measurements taken from Ref. [200].

Electrochemical characterization of fluorides. The electrochemistry of actinides in molten fluoride salts is studied in JRC Karlsruhe within the framework of research on properties of fuel candidates for MSR. The experimental set-up that was initially developed consists of a corrosion-resistant Inconel electrolyser inserted in a vertical furnace enabling a maximum working temperature of 1,200°C, built-in at the bottom of the same glove box as used for the synthesis of actinide fluorides (see synthesis of actinide fluorides, above). The lid of the electrolyser is equipped with ports for precise guiding and positioning of the electrodes attached to Inconel holders. The electrochemical cell consists of a glassy carbon crucible containing the measured molten salt and three electrodes. A working electrode is typically 0.5-1.0 mm metallic wire (such as W, Mo, Au, Ni, Ta, and Ag), a platinum wire (1 mm diameter) serves as a PtO₂/O²⁻- quasi-reference electrode, and a glassy carbon rod or a tungsten wire is connected as an auxiliary electrode. A photo describing the electrochemical set-up is shown in Figure 27, and details of the complete installation can be found in Ref. [205]. Since the electrode holders can be fully removed from the ports and the electrolyser can be closed gas-tight, the above-described HF gas line connected to the glove box allows for purification of liquid fluoride melts by bubbling pure HF gas at high temperatures.

FIG. 27. A photo of the electrolyser installed in JRC Karlsruhe with description of the most important parts of the set-up and electrochemical cell.

A method for purification of an LiF-CaF₂ (80.5-19.5 mol.%) eutectic carrier melt was established based on fluorination using HF gas bubbled directly into the melt through a nickel tube for 120 minutes at 850°C with a flow rate of 6 cm³/min. The procedure was proven sufficient to remove dissolved oxygen from the melt, as shown in Figure 28, which displays

the electrochemical response of oxygen in the melt before and after the purification. The details on the technique are published in Refs [205, 206].

FIG. 28. Linear sweep voltammetry of LiF-CaF₂ eutectic melt before and after purification by HF gas. Working electrode: Au wire (cross section (S) = 0.08 cm²), counter electrode: glassy carbon rod, reference electrode: Pt wire, scan rate: 10 mV/S, temperature: 850°C.

Electrochemical properties of thorium were studied in the melts LiF-CaF₂-ThF₄ (0.1-0.3 mol.% ThF₄) and LiF-ThF₄ (22.0 mol.% ThF₄) [205, 206]. Apart from verification of the published data on the electrochemical behaviour of thorium, a novel method usable for determination of activity coefficients of compounds dissolved in molten fluoride media using electrochemical techniques was proposed. The approach overcomes both the absence of a thermodynamic reference electrode in molten fluoride salts at high temperature and the usually unknown value of activity of free fluorine ions in the studied media. It is based on measurements of the difference between the electrode potentials of the studied compound and the least stable carrier melt constituent (melt decomposition potential). An experimental determination of this potential difference using cyclic voltammetry technique of the LiF-CaF₂-ThF₄ melt (0.18 mol.% ThF₄) is shown in Figure 29. The activity coefficient of the compound can be calculated from this value, assuming that the standard electrode potentials of the compound and the least stable solvent constituent are known or can be calculated, as well as the activity of the least stable solvent constituent. The activity coefficient of ThF₄ in the above-mentioned LiF-CaF₂-ThF₄ melt was determined to be 3.28×10^{-3} [205].

FIG. 29. Cyclic voltammetry of pure LiF-CaF₂ eutectic melt (thick line) and LiF-CaF₂-ThF₄ (0.18 mol.% ThF₄) melt showing graphical evaluation of the potential difference between ThF₄ and the least stable carrier melt constituent, LiF (working electrode: W, counter electrode: W, reference electrode: Pt, temperature: 850°C, scan rate: 100 mV/s).

Experimental studies on electrochemical and thermodynamic properties of other actinides in selected fluoride melts of interest are planned in JRC-Karlsruhe with an aim to create a database of properties important for fuel chemistry of MSR.

5.6.2.4. Properties at high temperature

As briefly addressed above, JRC has been significantly involved in experimental studies on properties at high temperature of molten fluoride salts in the last 15 years. A huge step towards achieving reliable results was accomplished by establishing a suitable and cost efficient encapsulation technique that prevents vaporization of samples during measurements at high temperature, and avoids chemical interaction between the material and the salt. Furthermore, the encapsulation technique must be selected for the purpose of the measurement and must be compatible with the measuring device. Two types of equipment have been suggested: one rather compact in size and based on laser welding of nickel capsules [207], and the other one using a firm stainless steel crucible with inert liner insets which can be made from nickel metal,

boron nitride (oxygen free), graphite, or any other inert material with respect to the sample [208]. Inner coatings are possible as well. Both types have been described in detail in earlier studies [208, 207] and their schematic drawings are shown in Figures 30 and 31.

FIG. 30. A schematic presentation and a picture of the nickel crucible designed for the encapsulation of fluoride and chloride salt samples. The wall thickness is ~0.2 mm. Middle picture: Metallography of the welded edge of the crucible.

FIG. 31. Schematic representation of the DSC crucible developed at JRC for measurements of halide samples.

Melting point determination and phase diagram investigations. One of the key properties of the MSR fuel is the melting temperature, which determines the lower limit of operating temperature of the reactor. Expertise in measuring melting points of halide salts has been developed at JRC and is available under the plan for Quality Assurance required by licensing authorities. Differential Scanning Calorimetry (DSC) is used for melting point determination by applying the encapsulation technique shown in Figure 31. Focus is given not only on melting point determination of pure fuel components or specific mixtures, but also on systematic investigations of key phase diagrams [191, 209, 210, 186], which provide solid bases for validation and extension of the JRCMSD thermodynamic database. Both examples are illustrated in the following figures: Figure 32 shows melting point measurement of the Molten Salt Fast Reactor (MSFR) fuel of the $\text{LiF-ThF}_4\text{-UF}_4$ (77.5-20-2.5 mol%) composition done within the framework of the EU project SAMOFAR, as published in the recent work by Tosolin et al. [211], determined as 828 K. The figure further summarizes the melting point determination of all three LiF , UF_4 and ThF_4 constituents as part of purity analysis of the starting materials. Figure 33 shows a recent study of the phase equilibrium points of the $\text{ThF}_4\text{-PuF}_3$ system, indicating three phase equilibria: (i) the liquidus temperature at which the first crystal precipitates upon cooling, (ii) the solidus temperature, represented by eutectic and peritectic melting, and (iii) the decomposition of the PuTh_2F_9 intermediate phase at high temperature.

FIG. 32. A DSC output of the melting point determination of LiF , UF_4 and ThF_4 end-members and the $\text{LiF-ThF}_4\text{-UF}_4$ (77.5-20-2.5 mol%) MSFR fuel.

FIG. 33. The assessed $\text{PuF}_3\text{-ThF}_4$ phase diagram. The points are phase equilibrium points obtained by DSC.

Heat capacity. Two techniques are used at JRC hot labs, the Differential Scanning Calorimetry (DSC) and Drop Calorimetry, for determining the heat capacity of molten salts. Both techniques provide reliable results and are at the same time suitable for fusion enthalpy or transition enthalpy evaluations, as discussed in the next subsection. The use of two independent methods is very useful for complementary determination of the fuel salt heat capacity required by the process of quality assurance. The focus is on three main domains at JRC:

- Systematic investigation of multi-component halide mixtures. Earlier studies from JRC [212, 213] show that heat capacity of a complex halide liquid solution does not follow ideal behaviour, that is, it cannot be simply estimated from the end-members (individual components forming the mixture) using the so-called Neumann-Kopp rule, but rather shows significant departures from this ideal trend. It has been further shown that the excess heat capacity is proportional to the cationic radius difference of mixing cations. The summary of the thus determined excess heat capacity for the liquid phase of the LiF-AlkF (Alk = Na, K, Rb, Cs) binary pairs is shown in Figure 34.
- Providing novel data to fill relatively large gaps existing for fundamental data on actinide halides, and re-evaluating existing data with current techniques, which meet the required standards. An example of novel measurement of the ThF₄ heat capacity for both solid and liquid phases utilizing complementary results obtained by DSC and Drop Calorimetry is given in Figure 35.
- Measurements of heat capacity of specific fuel composition. This study (and the upper one) is done under the scheme of quality assurance.

FIG. 34. Experimentally determined excess heat capacity as a function of molar ratio of mixed higher cation X(AlkF) for the LiF-AlkF binary liquid solutions: LiF-NaF (light blue); LiF-KF (red); LiF-RbF (green); LiF-CsF (orange). The experimental data are taken from Ref. [212].

FIG. 35. Heat capacity of solid and liquid phases of ThF₄ obtained by DSC and Drop Calorimetry; data from Ref. [214].

Fusion and mixing enthalpy measurements. As briefly mentioned in the above subsection, DSC and Drop Calorimetry techniques are excellent tools to determine enthalpies of phase transitions, including the fusion enthalpy. Figure 36 shows fusion enthalpy determination of the pure ThF₄ compound, directly obtained from enthalpy increment data of the solid and liquid phases. The thus determined value is in perfect agreement with a value measured by DSC, that is, 41.9 kJ/mol (not shown in the figure).

FIG. 36. Enthalpy increment data measured by Drop Calorimetry for ThF₄ solid and liquid phases as a function of temperature (T) (in °K); data are from Ref. [214].

Furthermore, the enthalpy of mixing of binary (and higher order) halide systems is determined using a DSC technique. This value is a measure of the liquid phase stability and gives valuable basis for phase equilibria modelling (described in “JRCMSD – Thermodynamic database”, below). Figure 37 gives an example of JRC’s earlier measurement of enthalpy of mixing of a binary LiF-ThF₄ system [209], indicating a relatively strong exothermic effect during mixing of the end-members.

FIG. 37. Mixing enthalpy of the LiF-ThF₄ binary system as determined by DSC; data are measurements taken from Ref. [209].

Vapour pressures. A Knudsen Effusion Mass Spectrometry (KEMS) device, shown in Figure 38, is used for determining the vaporization behaviour of MSR fuels and coolants. The whole assembly is placed in a glove box that is alpha tight and gamma shielded, thus allowing handling of radioactive materials, including irradiated materials.

KEMS is a powerful tool for the determination of the thermodynamic properties at high temperatures based on the equilibrium between condensed and gaseous phases. It can be used for identification of the gas composition and its vaporising behaviour as a function of temperature by means of the conversion of the detected spectral ion intensity data into the corresponding vapour pressures of the species in the gaseous phase. This method can further be used for determination of other thermodynamic properties, such as sublimation enthalpies, ionisation energies of gaseous species, activity coefficients or, as was the subject of this study, for the determination of the retention capacity of molten salt with respect to selected (Cs, I) fission products. This method is very reliable to determine boiling points of MSR fuels.

FIG. 38. Device for KEMS installed in alpha-tight glove box with external gamma shielding at JRC-Karlsruhe.

An example of measurement of vapour pressures of MSR fuel is given in Figure 39, which shows partial vapour pressures of the LiF-ThF₄-UF₄-PuF₃ (77.5-6.6-12.3-3.6 mol%) fuel mixture studied in the EU HORIZON2020 project SAMOFAR and published in Ref. [215]. It is evident from the figure that each fuel component evaporates at different intensities. In this case, the most volatile is the LiF matrix component, followed by UF₄ and ThF₄, and the least volatile is PuF₃. The partial vapour pressures of the homogeneously mixed fuel are thus proportional to their concentrations. The sum of the partial vapour pressures gives the total vapour pressure of the fuel, which is used to extrapolate the boiling point of the fuel, which in this case is determined as 1896±10 °K.

Fission product release – Retention capacity of the fuel. As mentioned above, the KEMS is a technique that can measure retention capacity of the MSR fuel for fission products. Understanding whether a fission product will dissolve in the fuel matrix or remain separated is very important, particularly in the case of a volatile fission product like caesium or iodine.

Recently, a series of experiments have been performed at JRC to understand the behaviour of these two major fission products and the effect of their chemical form on the volatility behaviour. Figure 40 shows the vaporization behaviour of CsF and CsI compounds mixed with LiF-ThF₄ eutectic solvent. It was found that CsF is completely dissolved in the fuel matrix, and thus its volatility is very low. However, as iodides are not very soluble in fluoride melts, the majority of CsI remained undissolved and consequently the volatility of CsI remained almost as high as for the pure compound. This is evident from the first release of CsI from the fuel, as indicated in the Figure 40.

FIG. 39. Partial and total vapour pressures for the mixture LiF-ThF₄-UF₄-PuF₃ (77.5-6.6-12.3-3.6 mol%). Relative uncertainty on vapour pressure measurements is 50%.

FIG. 40. Left: Volatility of CsF from the LiF-ThF₄ eutectic. Right: Volatility of CsI from the LiF-ThF₄ eutectic.

Thermal conductivity. The thermal conductivity measurements are carried out using the so-called LAF (LAsER Flash) device, shown in Figure 41. The device was fully designed and constructed at JRC-Karlsruhe, as described in detail in Ref. [216]. To measure radioactive elements, the LAF facility is installed in a glovebox, which is surrounded by lead shielding and equipped with two manipulators (the lead shielding allows thermal conductivity measurements of irradiated fuel materials). Two types of measurements are performed on a halide salt system: the solid phase to understand thermal transport through, for example, a solid crust layer precipitating on the inner walls of structural vessels; and the liquid phase.

The measurements of solid bulk are conducted on disks of small size with diameter of ca. 5mm and thickness of ca. 1mm, and it is important that the sample has as low porosity as possible. For this reason, most of the measurements are done on pre-molten samples which upon cooling solidify into a disc shape.

For the thermal conductivity measurements of liquid salts, a specially designed, laser-welded, hermetically closed crucible is used to keep the liquid phase in place during the measurements at high temperature.

FIG. 41. Left: Schematic representation of the LAF device used at JRC for measurements of thermal conductivity; right: LAF device installed in the hot labs.

Post Irradiation Examination. For its capability to work with molten salts containing actinides, JRC became key partner with the Dutch national program on molten salt irradiation, named the SALIENT project, led by NRG, Petten. The fuel for SALIENT01 irradiation has been synthesised at the JRC premises and in the near future part of the irradiated fuel will be transported back from NRG for post irradiation examination. This will include fuel

examination by KEMS to study volatility of the formed fission products and experiments performed by Transmission Electron Microscopy to investigate formation and size distribution of metallic precipitates. A benchmark to observations made to solid oxide fuels will be made. Currently JRC is involved in synthesis of fuel containing plutonium ($75\text{LiF}-18.7\text{ThF}_4-6\text{UF}_x-0.30\text{PuF}_3$ mol% composition) for SALIENT03 irradiation, which will be hermetically sealed by qualified welding using alloys based on nickel, such as US Hastelloy-N and Chinese GH3535 alloys. More information about the Dutch program on irradiation of molten salts is given in Subsections 5.10.1.3, 5.10.1.4, 5.10.1.6, and 5.6.3.

JRCMSD – Thermodynamic database. JRC is the owner of an extensive thermodynamic database (JRCMSD) which describes the key MSR systems studied around the world, including chloride and fluoride salt media. The database is continuously developed in collaboration with other partners (e.g. TU Delft) by novel assessments and data reviews addressing not only fuel behaviour in the so-called fresh state, but also fission product behaviour and corrosion interactions. The database has been recently shared with members of the TAF-ID project (international project on development of nuclear thermodynamic database), organized by NEA-OECD. A summary of the described binary systems from the JRCMSD is given in the Figure 42. The database provides the complete thermodynamic description of the following key systems (with main references included):

- LiF-NaF-BeF₂-PuF₃ system: Fuel for actinide burner with BeF₂ as component [217];
- LiF-NaF-KF-RbF-CsF-LaF₃-PuF₃ system: BeF₂-free fuel for actinide burner with consideration of KF and RbF as matrix alternatives and CsF and LaF₃ as fission product representatives [218];
- LiF-NaF-BeF₂-UF₄-ThF₄ system: Fuel for breeder reactor using molten salt with NaF as alternative candidate for the matrix component [219];
- LiF-NaF-UF₃-UF₄ system: Fuel system showing the influence of UF₃ content on, for example, melting behaviour of the fuel. UF₃ is considered a redox control additive [220];
- LiF-ThF₄-CeF₃-PuF₃ system: Fuel for non-moderated breeder reactor using molten salt with CeF₃ as representative of a fission product (and as proxy to PuF₃ at the early stage of the database development) [210];
- LiF-CsF-ThF₄-LiI-CsI-ThI₄ system: Fuel for key MSFR system with addition of fission products CsF and CsI [221];
- NaCl-MgCl₂-UCl₃-PuCl₃ system: Breeder and burner concept utilizing chloride salts [222].

FIG. 42. Status of binary systems in the JRCMSD thermodynamic database.

The database is a very good tool to predict some key properties of the MSR fuel, such as melting point, chemical nature of precipitates, heat capacity, vapour pressure, heats of fusion or boiling points. It was also found as a very useful tool for optimizing the composition of an MSR fuel mainly due to the multi-component nature of the fuel, and the associated database ability to simulate properties of any composition needed. Figure 43 shows examples of calculated melting behaviour (the liquidus projection) of two selected systems: The LiF-ThF₄-PuF₃ and the NaCl-UCl₃-PuCl₃ systems.

FIG. 43. The calculated liquidus projection of the $\text{LiF-ThF}_4\text{-PuF}_3$ and $\text{NaCl-UCl}_3\text{-PuCl}_3$ systems (left and right, respectively).

5.6.3. Component and technology development

The High Flux Reactor (HFR) is a Material Test Reactor located in Petten, owned by the European Union and operated by NRG. Since 1961, the HFR has been employed to conduct irradiation experiments to investigate the behaviour of both structural materials and fuel exposed to high thermal neutronic flux. Recently (2017-2019), it has been used to irradiate molten salt within the frame of a bilateral collaboration between the JRC and NRG. The experiment conducted and the other experiment under preparation are called SALIENT-01 and SALIENT-03, respectively. Since the deployment of the first experiment (SALIENT-01; see Subsection 5.10.1.6 on this experiment), which used capsules made of graphite containing a mix of salts based on fluoride ($78\text{LiF-}22\text{ThF}_4$) [223], some experience was gained and new challenges resulting from handling of MSR fuel under irradiation were successfully tackled.

Although there is good and well documented compatibility between molten salt based on fluoride and nuclear grade graphite, the second experiment (SALIENT-03) was conducted encapsulating molten salt ($75\text{LiF-}18.7\text{ThF}_4\text{-}6\text{UF}_x\text{-}0.30\text{PuF}_3$) in alloys based on nickel (Hastelloy-N and GH3535). Such choice allowed a more precise understanding of the fission products released and trapped in molten salt, as well as to study the interaction between the salt and the metallic structural material [224]. In general, the objectives of SALIENT-03 are:

- (a) To study the in-pile corrosion of the MSR containment material Hastelloy-N and an equivalent alloy (GH3535) when they are in contact with molten fluoride fuel salt, as well as:
 - (i) In a strong, stable temperature gradient ($> 100^\circ\text{C}$);
 - (ii) As a function of the redox potential of the salt;
 - (iii) As a function of the concentration of the corrosion products in the salt;
 - (iv) In the presence of corrosive fission products generated during irradiation (notably tellurium).
- (b) To continue and improve the study of fission product behaviour in molten fluoride salts representative of MSRs for breeding thorium as started in SALIENT-01.

The design of SALIENT-03 has been tailored to achieve the following goals:

- A fission burnup of at least 2% FIHMA (Fission of Initial Heavy Metal Atoms) to be able to produce enough amount of fission products dissolved in the salt to be analysed during the post irradiation examination phase.
- A constant irradiation temperature of the capsules containing the molten salt of at least 700°C maintained for 10000 hours to allow the significant development of the corrosion process, and a simple correlation between corrosion rate and operating temperature.
- Irradiate 5 capsules containing a slightly modified composition of the salt to investigate the influence of redox potential and corrosion products in the corrosion process of the capsules containing it.

- Develop a sophisticated quality system based on redundant checks of the composition of the salt to assure its purity. The purity of the salt is a key issue to understand its behaviour under irradiation.
- Addition of electric heaters for operation when the reactor is not operating, to avoid radiolytic production of F₂ gas and an associated reduced state of the salt samples [225], [226].

The SALIENT-03 experiment has been equipped with instrumentation to collect as much information as possible during in-pile irradiation:

- 43 thermocouples to map the temperature distribution of the irradiated capsules with the great accuracy.
- 15 flux monitors (3 per capsule) to check the neutron fluence to which the salt has been exposed.
- A pressure transducer connected to the first capsule to check online the built-in pressure due to the generation and release of gaseous fission products during irradiation.
- 3 electrodes connected to the first capsule to check online the variation of redox potential during irradiation.

The information acquired during the irradiation will be complemented with a comprehensive post-irradiation examination of the capsules and the salt contained in them.

The SALIENT-03 contains five capsules. Four of them made of Hastelloy-N and the fifth made of GH3535. The composition of the salt inside each capsule is slightly different to study the corrosion of the most promising alloys based on nickel under irradiation when exposed to different redox potentials and different CrF₃ concentrations. Four of the five capsules irradiated have the same geometry. The fifth is longer than the others, thus allowing buoyancy to increase in the salt and allowing measuring the temperature in the centre of the capsule and not only on its external wall. Figure 44 shows a picture of the capsules with their safety containments.

FIG. 44. Section of the SALIENT-03 experiment.

The different components of the salts have been synthesized at JRC Karlsruhe. Tests for the procedure of filling the capsules with salt and qualified welding were performed as well. The design of the experiment has been completed, and the fabrication of the capsules and sample holder is ongoing. The SALIENT-03 irradiation is scheduled to start in the second quarter of 2021.

5.7. R&D ACTIVITIES IN FRANCE

5.7.1. Introduction and main achievements

Since more than 20 years, the French National Centre for Scientific Research (Centre National de la Recherche Scientifique (CNRS)), with national and European partners has carried out R&D developments on the molten salt reactor concept called the Molten Salt Fast Reactor (MSFR). The Generation IV International Forum (GIF) selected the MSFR (representative of

reactor with molten salt as fuel and coolant) and the AHTR (representative of reactor with molten salt as coolant only) due to their promising safety and design characteristics [227]. The so-called ‘reference MSFR’ design is a 3,000 MW(th) reactor based on a fluoride fuel salt volume of 18 m³ and operated at a mean fuel salt temperature of 700°C [228]. New research activities started in 2018 also on small modular versions of the MSFR (s-MSFR), to be operated as a breeder in the U/Pu fuel cycle or as an actinide burner and thus using a chloride salt. These R&D activities are currently expanded with a growing national interest in France to assess the ability of these MSFR systems to satisfy the objectives of Generation IV reactors [229] in terms of sustainability, resource saving (closed fuel cycle, no uranium enrichment), safety, waste management (actinide burner) and non-proliferation.

From 2009 to 2019, the public research on MSFR in France was mainly conducted by CNRS and French universities, with the participation of the Institute for Radiological Protection and Nuclear Safety (Institut de Radioprotection et de Sûreté Nucléaire (IRSN)) and industrial partners (CORYS, EdF, Framatome, Orano), and recently of the French Alternative Energies and Atomic Energy Commission (Commissariat à l’Energie Atomique (CEA)). It was supported by French research programs (PACEN, NEEDS, MSFR Master Project of CNRS/IN2P3) and European programs (EVOL, Safety Assessment of the Molten Salt Fast Reactor (SAMOFAR), and Severe Accident Modelling and Safety Assessment for Fluid-fuel Energy Reactors (SAMOSAFER)). The manpower varied from 5 to 15 researchers per year during this period, most of them working on basic data acquisition and physical phenomena modelling.

The R&D activities in France focused on the MSFR concept and are related to the following topics as detailed in the following subsections:

- Calculations of material composition, nuclei evolution, and burnup associated with MSRs: development and validation of the REM neutronic code, and studies and optimization of different kinds of MSRs (fluoride and chlorides salts, fast and thermal spectrum, a cycle based on Th-²³³U or on ²³⁸U-Pu, or versions for burning actinides).
- Simulation studies during normal and accident conditions: development of two simulation tools for MSRs, namely, the TFM-OF code, a coupled Computational Fluid Dynamics (CFD)-neutronics code that accounts for the delayed neutron precursor motion in the circulating liquid fuel and developed by CNRS for transient analysis; and the LiCore power plant simulator under development by CNRS and CORYS.
- Safety-by-design for MSRs: Safety evaluation of the ‘reference’ MSFR (a 3,000 MW(t) MSFR based on the Th-U cycle) to achieve safety-by-design for a large MSFR, with the plan to apply the same approach for the S-MSFR in the coming years.
- Measurement and study of the basic chemical properties of molten salts: The main French experimental activities on MSRs are driven by the Laboratory of Subatomic Physics & Cosmology (LPSC Grenoble) with the FFFER (Forced Fluoride Flow for Experimental Research - devoted to test gas dispersion and separation in circulating salts) and the SWATH (Salt at WALL: Thermal excHanges - to validate heat transfer models in presence of a solidified layer of salt in contact with a circulating liquid salt) experimental loops using liquid LiF-NaF-KF eutectic (FLiNaK) at 550/750°C, and by the Laboratory Irène Joliot-Curie of the Physics of the two infinities (IJC Lab Orsay) for the chemical and material experiments, as described in Subsections 5.7.4 and 5.7.5. Two other laboratories (LGC-

Toulouse and CEMHTI Orleans) are not presently supported anymore in the MSR field but have experimental expertise if necessary.

- Corrosion studies: In particular, the influence of the salt redox potential on corrosion. These studies have been carried out with molten salts in contact with Hastelloy C and Steel AISI 304.

Concerning the prospects of these R&D activities, the work will continue on the same topics during the SAMOSAFER European project (2019-2023) focusing on the ‘reference MSFR’, the fluoride breeder version using the thorium fuel cycle. Other studies started since 2019 on alternative versions of the MSFR based on a chloride salt and designed as breeders in the U/Pu fuel cycle or as actinide burners. Some of these alternative versions are small modular reactors (SMRs). All these versions (the reference and alternative ones) are characterized by a fast neutron spectrum and by an integrated design for the fuel circuit. These activities are done in the frame of national collaborations between academic (CNRS, Grenoble Institute of Technology, IMT Atlantique) and industrial (Framatome, Orano, CORYS, EDF) partners, together with CEA and IRSN.

5.7.2. Reactor physics

5.7.2.1. *Simulation of material evolution during operation*

The MSFR team based at the CNRS/LPSC laboratory of Grenoble is developing, updating and validating the REM neutronic code for more than 20 years [230, 231]. This code has been used to perform various MSR studies and optimizations of the MSFR concept based on the evolution of material composition during irradiation or in storage, including burnup studies of different kinds of MSFR (fluoride and chlorides salts, fast and thermal spectrum, and versions of MSRs as breeders and actinide burners using a cycle based on Th-²³³U [232] and based on ²³⁸U-Pu). A neutronic benchmark [233] has been carried out in the frame of the EVOL European project. Finally these burnup calculations have been used for applications such as proliferation evaluations [234, 235] and radioprotection issues in the frame of the work-package 5 of the SAMOFAR project between CNRS/IPN Orsay, CNRS/LPSC and CEA/Institute of research into the fundamental laws of the Universe (IRFU).

Some dedicated studies were performed at CNRS/SUBATECH to evaluate the decay heat produced by a MSFR and to assess the potential needs of new nuclear data for such a Generation IV concept. The decay heat is calculated by combining reactor simulations to estimate the fuel inventory with nuclear data (decay properties of fission products and actinides, cross sections and fission yields) as inputs. The codes currently used are mainly validated for boiling water reactors (BWRs)/pressurized water reactors (PWRs) in the U/Pu cycle, but they are not yet validated for Generation IV reactor concepts. Moreover, some fission products in the decay data libraries have decay schemes biased due to the Pandemonium effect. This Pandemonium effect comes from the low efficiency of Germanium detectors at high energy, resulting in an overestimation of the β^- contribution and an underestimation of the γ contribution in the decay heat. The SERPENT2 code was used to carry out a simulation of the MSFR reactor core operated with a fluoride salt, based on the ²³²Th-²³³U cycle. The aim was to identify the main nuclei contributors to decay heat for different cooling times and to see if some important fission products are also biased by the Pandemonium effect and need to be re-measured with an alternative experimental technique based on the Total Absorption Spectroscopy method. A preliminary list of 10 potential Pandemonium nuclei was established

[236], and will be extended/compared to cases with chloride salts and the ^{238}U - ^{239}Pu cycle, before being discussed with the TAS collaboration (Subatech, Univ. of Surrey, IFIC Valencia) for a potential experimental proposal on an ISOL (Isotope Separation On-Line) facility.

In addition, some work is also foreseen (2020-2024) in uncertainty calculations of the MSFR's decay heat, especially to determine the impact of nuclear data by simultaneously using approaches based on the Total-Monte Carlo method (TMC) and the perturbation theory.

5.7.2.2. *Simulation tool for transient analysis - The TFM-OF code-*

The MSFR, as a reactor with circulating liquid fuel, requires a new definition of its safety and operating procedures. The intrinsic core stability is guaranteed by its excellent negative feedback coefficients. Thanks to this stability, the power production may be driven only by the heat extraction, resulting in a very interesting flexibility to follow the electric grid's load; control rods in the core may then not be required, which eliminates a classical accident initiator. Since the heat is produced directly in the salt that circulates in the fuel circuit, the negative feedback coefficients act very rapidly to stabilize the core, unlike reactors with solid fuel. The fuel salt itself is cooled in the heat exchangers. Due to these specificities, dedicated tools are being developed to simulate the reactor's behaviour for normal (e.g. load following) and accident (e.g. reactivity insertion) conditions, to develop, optimize, and assess the MSFR operation procedures. The reactor modelling requires specific treatment to consider the phenomena associated with the liquid fuel circulation as detailed next.

The study of MSR cores in such conditions requires a code coupling the neutronic and thermohydraulic evolutions. Accordingly, two tools for transient simulation have been developed at CNRS: the TFM-OpenFOAM 3D-coupled code, and a simulator of an MSR power plant based on the LiCore core model (see Subsection 5.7.3.1).

Important aspects of such systems regarding the coupling are the delayed neutron precursor convection and a complex flow pattern in the core cavity. Thus, a multiphysics tool, called Transient Fission Matrix (TFM) [237, 187], that is more generic than those existing for reactors with solid fuel, has been developed by coupling CFD and a green function approach based on Monte Carlo calculations. The CFD allows to solve the Reynolds Average Navier Stokes (RANS) equation and provides a 3D flow description in the core. The TFM approach is a time-dependent version of the fission matrices characterizing the transport of a neutron from its birth position to its death/fission position. This includes the prompt as well as the delayed neutrons whatever their location in the fuel circuit are (in the critical zone or in the recirculation/cooling sectors). Then, using a technique of power iteration, all the generations of prompt neutrons are reconstructed, and finally the reactor fission distribution, that accounts for the precursor transport, is obtained. The description of the neutronics approach and this approach's coupling to the OpenFOAM is available in Refs [187, 69].

The TFM-OF may be used to calculate various normal (start-up and load following procedures) and abnormal transients (reactivity insertion, loss of fuel flow, loss of heat sink, and low power overcooling accident). Depending on the calculation options and mainly the refinement of the meshing, the calculation time for a given transient varies from several seconds (real time calculation) to one week. Finally, the TFM-OF code developed initially for MSFR studies was also used for PWR calculations [187], and recently was extended to perform transient calculations of sodium fast reactors [238, 239, 240, 241] and of research reactors [242].

5.7.3. System behaviour and safety evaluation

As mentioned above, molten salt reactors whose fuel is in liquid form are indeed flexible in terms of operation (such as the capability to follow the grid's load) and design choices (such as fuel composition, power level...), and they are very different in terms of design, operation and safety approach compared to reactors with solid fuel. Such reactors require a new definition of their operating procedures.

5.7.3.1. *The simulator for the MSFR power plant: the LiCore code (CNRS / CORYS collaboration)*

Dedicated developments and studies have been performed in the frame of the European SAMOFAR project of Horizon2020 and in parallel in France involving CNRS, CORYS and Framatome on the code called LiCore. This code is a power plant simulator based on basic principles and adapted for MSRs.

The MSFR plant is composed of three circuits: the fuel circuit, the intermediate circuit, and the power conversion circuit. The fuel circuit, defined as the circuit that contains the fuel salt during power generation, includes the core cavity and the recirculation-cooling loops or sectors, which are mainly comprised by the inlet and outlet pipes, pumps, and fuel heat exchangers. The neutronic model LiCore, at the centre of the simulator, corresponds to an improved model of point kinetics to account for the specificities of an MSR, notably the circulation of the delayed neutron precursors out of the core. Coupled to a piston model for the fuel motion in the core, this code can perform calculations faster than real time to simulate the behaviour of the fuel circuit. A simplified model of the intermediate circuit allows to run parametric studies of the MSFR fuel circuit during normal and accident conditions. Consistency of the results provided by LiCore code with the 3D coupled neutronic thermohydraulic TFM-OpenFOAM code has been checked [187, 243].

FIG. 45. Global scheme of the LiCore-ALICES power plant simulator.

Finally, CNRS (developing the LiCore simulator) is collaborating since 2017 with the CORYS company, which is a subsidiary company of Framatome that develops simulators for trains and nuclear power plants (NPPs). The LiCore code has been integrated successfully in ALICES (see Figure 45), the integrated simulation toolset designed by CORYS for the development, maintenance, and operation of major simulators, such as power plant simulators. Additional modules are being added to fully simulate the intermediate and energy conversion circuits. The idea is to add a simulation of the intermediate and energy conversion circuits. This integrated version allows to study the whole MSFR plant (see Figure 46), thus helping to define the operating procedures of the reactor. The next steps for developing this power plant simulator will be the addition of control-command devices and the improvement of the modelling of the components, such as the turbine.

5.7.3.2. Normal operation procedures

Studies have been performed with the LiCore and the TFM-OF codes, together with expert judgement, to propose some operation procedures such as the startup [244] and the procedures for following the electric load [69]. This procedure definition continues in the frame of the SAMOSAFER European Project.

FIG. 46. Main screen of the LiCore-ALICES power plant simulator for the MSFR.

5.7.3.3. Geometry optimization of the fuel circuit

A static system code focused on the MSFR's fuel circuit and connected to genetic algorithms has also been developed at CNRS/LPSC. This code allows optimization studies of this circuit. The objective is to obtain a globally optimized geometry of the fuel circuit by setting different parameters and constraints, instead of separately optimizing the various components (such as heat exchangers and pipes). This is mandatory since some constraints cover several components (for example, the fuel salt-coolant volume). A configuration that best satisfies these constraints can then be sought by adjusting a list of variable design parameters.

This code thus describes the system as an assembly of components, each component having specific properties (physical properties) or parameters (such as length, hydraulic diameter, and power), see Figure 47. It sets a set of constraints on each parameter (minimum and maximum values) with a contribution to the figure of merit of the system. Then, a genetic algorithm optimizes the global figure of merit of the system by changing the parameters. An example of application is the optimization of the fuel circuit with a competition for the total fuel salt volume to be reduced to lower the fissile inventory but to keep it large enough to reduce the pressure drops in the pipes and heat exchangers.

FIG. 47. Sizing parameters for the fuel circuit of the MSFR.

Two types of heat exchangers have been considered following some preliminary assessments performed by Framatome: corrugated heat exchangers using Hastelloy, plate heat exchangers using silicon carbide (SiC), and channel heat exchangers. Calculations have been performed, for instance, with given inlet temperatures of the intermediate fluid in the heat exchangers and fuel temperature in the core, and with given mass flows of the salts in the intermediate and fuel circuits. The results are the geometric parameters of the heat exchangers (plate heat exchangers are better than channel heat exchangers) and that the main risk is due to the lowest fuel temperature (freezing). Applications include the main temperatures (criticality temperature and wall temperatures) and hydraulic characteristics as functions of the core specific power, ranging from 50 to 600 MW/m³.

5.7.4. Safety approach

A safety approach dedicated to the MSFR or more comprehensively for reactors with fuel that is circulating in liquid state is being developed. Some preliminary steps have been studied in the frame of the EVOL project as well as in M. Brovchenko's PhD thesis [245]. The approach for this development has been defined in the SAMOFAR project in a task led by the IRSN (France), and involving mainly the CNRS (France), Framatome (France) and Politecnico di Torino (POLITO, Italy). This assessment methodology aims at reaching a safety that is "built-in" and not "added-on" by applying it at the earliest stages of design. The approach is based on the Integrated Safety Assessment Methodology (ISAM) developed by the Risk and Safety Working Group of the Generation IV International Forum [246], coupled to usual risk analysis methods such as functional method analysis, master logic diagram, and line of defence method [247, 248].

Based on the new integrated design of the MSFR plant, abnormal situations of the fuel circuit and the emergency draining system have been identified and classified according to their initiating event (i.e., according to the phenomena involved), based on the following sources:

- The analysis of the accident types identified for currently operating pressurized water reactors (PWRs);
- Deterministic calculations, such as criticality and thermal studies [244, 249], and calculations of transient and accident scenarios using multiphysics calculation tools [187, 69];
- Preliminary risk analyses taking into account design optimizations, together with the identification of the associated initiating events [250, 247, 248].

The application of the assessment methodology combining deterministic and probabilistic tools has been launched during the SAMOFAR European project, for example, to identify a reference severe accident of the MSFR, and to propose and evaluate possible confinement barriers and lines of defence. This work will continue in the frame of the SAMOSAFER European project (2019-2023).

5.7.4.1. Abnormal transient calculations

As already mentioned, various MSFR transient studies have already been calculated with the TFM-OF code, including parametric studies of overcooling and reactivity insertion transients. A parametric study of an overcooling transient is illustrated in Figure 48 [69].

FIG. 48. Distribution of the fuel salt temperature at $t = 0$ (left), and its time variation $T(t) - T(0)$ for a 100 MW to 3 GW overcooling transient.

These transient calculations have highlighted the excellent stability of the MSFR core even in the case of a violent and quick perturbation. Parametric transient studies (such as the overcooling event at low power level illustrated above) performed up to prompt critical regime [69] have also demonstrated that no cliff-edge effect occurs when this regime is reached, that

is, no sudden violent behaviour is observed for the MSFR. This behaviour is very important regarding safety considerations and analyses.

5.7.4.2. Fuel and coolant chemistry

These studies rely on the chemistry of fission products and actinides in the fuel salt. The fuel salt selected at the end of the EVOL project for the MSFR concept was LiF-ThF₄-UF₄ (77-19-4 mol%). The main experiments carried out at IPN in Orsay (now IJC Lab) were focused on the solvent salt LiF-ThF₄ (77-23 mol%) to study the behaviour of uranium and fission products in this salt [251, 252]. The fission products considered were iodide, zirconium and lanthanides. The possibility of extracting iodide by forming iodine gas by fluorination with an efficiency higher than 98% was shown. Zirconium chemistry was examined, and it appears Zr(IV) is the only one soluble oxidation state stable in the molten salt; this is in disagreement with thermodynamic calculations which indicate a stability domain for ZrF₂ [253]. The chemistry of uranium was particularly well studied because (a) uranium is the fissile material of the MSFR concept, and (b) it can exist as two soluble states U(IV) and U(III), and this characteristic is used to control the redox potential of the fuel salt to avoid the corrosion of the structural materials. Subsection 5.7.5 further discusses this point.

In presence of less than 2 parts per million (ppm) of O₂ in the environment in equilibrium with the salt, an oxidation of iodide (I⁻) to iodine (I₂) is observed with formation of stable thorium oxy-fluoride (ThOF₂), whose solubility has to be determined. The high stability of this compound leads to increasing the oxidizing power of O₂. When UF₄ is added in the molten salt without UF₃, a decrease of its concentration with time occurs, which is due to the chemical oxidation of UF₄ by O₂, which in turn produce soluble UO₂F₂. The solubility of the oxy-fluorides UO₂F₂ and ThOF₂ has not yet been evaluated in order to know from what amount they will precipitate.

The thermodynamic diagram of uranium in LiF-ThF₄ was drawn based on thermodynamic data and experimental results (see Figure 49). In this figure, $E(V)$ represents the potential of the salt and $pa(\text{Li}_2\text{O})$ the cologarithm of the activity of Li₂O, which represents the level of oxides in the salt. This kind of diagram is usually used to evaluate the thermodynamic stability of the salts [142]. The stability domain of each compound is presented by areas limited by straight lines. The red lines correspond here to the Th compounds and the green ones to the U compounds.

FIG. 49. Thermodynamic diagram of stability of uranium in LiF-ThF₄ molten salt at a temperature of 650°C ($P(\text{F}_2) = 1\text{atm}$, $a(\text{F}^-) = 1$, $[\text{UF}_4] = 2.5\text{ mol\%}$) [252].

Some fundamental data were determined from these studies, such as activity coefficients and diffusion coefficients in LiF-ThF₄. These data are gathered in Table 14.

TABLE 14. DATA EXPERIMENTALLY DETERMINED IN LIF-THF₄ AT 600 AND 650°C

	<i>T</i> (°C)	UF ₄	UF ₃	ThF ₄	KI	ZrF ₄
Activity coefficient, <i>lg γ</i>	600	-4.34	-0.54	-1.98	ND	ND
	650	-4.05	-0.52	-1.96	-1.63	-4.33
Diffusion coefficient, <i>D</i> (cm ² /s)	650	1.85 × 10 ⁻⁶	2.5 × 10 ⁻⁶	ND	ND	ND

ND = Not determined.

The main steps of the processing scheme of the MSFR spent fuel are displayed in Figure 50. The processing is composed of two parts: an on-line physical processing done by a bubbling gas injected in the fuel salt (in green), and an off-line chemical processing, which consists in removing some tens (10 to 40) litres of the fuel salt every day for a batch processing of several chemical reactions based on the redox and acido-basic properties of elements in the fuel salt. Since the different elementary steps involved in off-line processing rely on the chemical properties of the elements composing the fuel salt, a considerable amount of basic data is required to calculate each step efficiency, as discussed below. The red rectangles correspond to the storage steps in Figure 50.

FIG. 50. Main steps of the chemical processing scheme of the MSFR fuel salt, including the physical (bubbling) processing on the left top (gas) [254].

Unlike the MSBR project studied at ORNL in the 1960s [255, 256], based on a thermal neutron spectrum, the removal of ²³³Pa is not an issue in the MSFR concept thanks to its fast neutron spectrum. Indeed, due to the high neutron captures by ²³³Pa and fission products (FPs) in a thermal neutron spectrum, the required rate of fuel salt processing was estimated at about 4,000 l/day for the MSBR. The fast neutron spectrum of the MSFR concept allows a much slower processing rate of 10 to 40 l/day and no ²³³Pa removal is required for the MSFR, Pa being managed with the other actinides in the chemical processing steps.

(i) On-line processing: Helium bubbling

This part of the processing has two main objectives: the removal of gaseous fission products Xe and Kr, and the removal of part of the noble metals produced in the core when they are in their metallic (solid) states. This process, based on flotation process, has already been studied by the ORNL. An amount of 15% of the gaseous fission products having a very short lifetime will decrease in the salt before extraction. The other gaseous FPs (85%) have lifetimes that are long enough to allow their extraction and treatment.

(ii) Off-line processing

The fluorination step has been initially studied at ORNL and the results obtained have been adapted for the MSFR processing. The objective is to remove the elements with high gaseous oxidation states, such as Nb, Te, I, Mo, Cr and Tc for the fission products, and U, Pu and Np for the actinides. This technique consists in oxidizing all the chemical elements contained in the salt to their higher oxidation states to produce gaseous elements, which are naturally separated from the salt.

A part of plutonium can also be oxidized in certain conditions [257]. The extraction by fluorination is possible by using small (around 100 μm) liquid salt droplets falling in a 50% F_2 atmosphere at a temperature of 550/650°C. Up to 90% of Pu was removed from the salt falling in a 1.3 m high fluorinator, and this efficiency is expected to be improved at higher temperatures and with smaller drop diameters. The Pu extraction that required crushing of solid salt was not feasible at a large scale in the 1970s. Today, granulation in a diameter range corresponding to that of the MSFR has reached an industrial stage for liquids at high temperature. This industrial technology can be adapted to the formation of molten fluoride droplets falling in a quasi-stagnant fluorine atmosphere.

The gaseous elements are then adsorbed on NaF traps. This adsorption reaction depends on the temperature of the traps. For instance, at 400°C, the adsorption of fission products (such as molybdenum, tantalum, niobium, tellurium, and fluorides) occurs, but adsorption of UF_6 and NpF_6 does not happen. The adsorption of these two actinides occurs when the gas flows through a second trap heated at 25°C. The gas flows through several NaF traps heated at a given temperature to adsorb specific fission products or actinides and to perform their separation. In a second stage, the desorption of the elements happens by heating the traps at higher temperature. Fission products are reduced by hydrogen and are managed as waste. The objective of the last step is to reduce the actinides extracted by fluorination by using hydrogen gas in order to re-introduce them back in the fuel salt. Only a proportion of UF_4 is re-introduced, the remaining amount extracted being stored to feed new reactors since the MSFR is a breeder concept. Finally, the remaining gaseous elements removed by fluorination are also reduced to their solid state with hydrogen gas before storage. It appears that all the fission products are retained by absorption on the first trap (at 400°C), and all the actinides are absorbed on the second trap (at 25°C).

Concerning the reductive extractions step, a large amount of results from ORNL can be used to define the MSFR processing concerning the steps of fluorination and helium bubbling previously described. As a consequence, in the frame of the first period of the EVOL project, the R&D activities on the processing scheme focused on the separation of actinides (An) and lanthanides (Ln) since no data were available for the calculation of the associated efficiencies. This separation was expected to be performed in two steps based on reductive extraction, with a 'first reductive extraction' dedicated to actinides, and a 'second reductive extraction' for lanthanides. The reductive extraction is done through the contact between two liquid phases: the molten salt containing the soluble actinides and fission products that have to be extracted, and a liquid metal made of liquid Bi, which contains Li. It was demonstrated that the amount of Li is the main parameter to control the selectivity and efficiency of An/Ln separation [258].

Reductive extraction requires the use of a Bi-Li liquid mixture of high purity. This mixture is prepared by cathodic electrolysis in the LiF-LiCl molten salt on a purified liquid electrode of bismuth [259]. Reductive extraction tests were performed at lab-scale in LiF-ThF₄ containing

U and Nd to simulate An and Ln, respectively [260], in contact with Bi-Li (10 mol%). The efficiency measured was very low because it was shown that a solid alloy of Th_3Bi_4 and BiTh was formed at the interface liquid metal/molten salt which blocks the transfer of U and Nd from the molten salt to the liquid metal.

The final step concerns the back extraction of actinides and lanthanides. The back extraction of actinides is naturally achieved in the cleaned LiF-ThF₄ salt by contact with the Bi pool containing actinides. This salt mixture is re-introduced in the reactor core after controlling the redox potential and adjusting the amount of fertile and fissile material (introduction of ThF₄ and UF₄, if necessary). The back extraction of lanthanides is accomplished in an inactive molten salt, possibly LiF-LiCl. The lanthanides are precipitated in oxide form by bubbling an inert gas saturated with a mixture of steam (pure water vapor) and other gases at a given temperature. In this way, the lanthanide oxides precipitation is attained, with formation of HCl or HF gas [261]. The important aspect of this approach is to limit the volumes of effluents.

Evaluation of the decay heat in the reprocessing plant and shielding requirement. During the SAMOFAR project, an evaluation of the safety of the reprocessing plant has been done by CNRS and CEA/Irfu [262]. The first step consisted in providing the amount of each isotope in all the places of the chemical plant. To do this evaluation, it was necessary to know the initial inventory of isotopes and to introduce the efficiency of extraction of all the elements in all the chemical steps. That was done considering both experimental results and thermodynamic data. Figure 51 gives the final scheme, where RE stands for "Reductive Extraction," An for actinides, FP for fission products, and Ox for oxides. "ONE DAY" in the dotted blue rectangle means that the processing takes 1 day.

FIG. 51. Decay heat in each step of the processing scheme and in the storage zone.

This work highlighted the necessity of defining one more step in the reprocessing scheme, which is indicated by "cooling phase." This step consists in waiting for one day for the decay heat in the batch to decrease before starting the reprocessing. During this period, ²³³Th decays to ²³³Pa and the heat decreases by 157 kW.

By using the information in Figure 51, the shielding requirement in the chemical plant was evaluated to be around 3 meters of concrete in the Th-breeder reference version of the MSFR.

Chemical experiment facilities for salts containing actinides in Orsay. Several glove boxes are installed at CNRS/IPN Orsay to perform experiments in active and inactive conditions. The glove boxes are under inert gas, in depressurized conditions, or in both conditions. Work can be carried out with about 100g of ThF₄ and UF₄. Other equipment is available such as a TGADSC device (Thermo Gravimetric Analysis-Differential Scanning Calorimetry) to measure melting points, and an ICP-AES (Inductively Coupled Plasma-Atomic Emission Spectrometry) for elemental analysis in molten salts, X-ray diffraction, and Scanning Electron Microscopy.

5.7.5. Materials performance

5.7.5.1. Influence of the redox potential on the materials corrosion

The redox potential of the salt is the main parameter to control the corrosion of the structural material. As shown in Figure 49, the fuel redox potential can be controlled by regulating the ratio $[UF_4]/[UF_3]$ using the Nernst relation:

$$E_{salt} = E^{\circ}_{UF_4/UF_3} + 2.3 \frac{RT}{F} \lg \frac{[UF_4]}{[UF_3]}$$

In this relation, R is the ideal gas constant, T is the temperature (K), and F is the Faraday constant.

This potential can be adjusted by adding U metal in the molten salt containing UF_4 . In this case, it was shown that the following chemical reaction occurs: $3UF_4 + U \rightarrow 4UF_3$, which is a very fast and total reaction. Therefore, the mass of uranium metal added is directly related to the amount of UF_3 produced and the ratio $[UF_4]/[UF_3]$ can be adjusted in this way.

To test the influence of the redox potential on the corrosion, some tests were carried out at $600^{\circ}C$ during 360 hours with and without potential control on two alloys: Hastelloy C276 (Ni-based alloy) and AISI 304 (Fe-based alloy), highlighting the very important and efficient role of redox potential control to avoid corrosion in both cases [263].

Experiments in chloride media also are under way. Thermodynamic calculations evidence the risk of formation of plutonium oxide and chlorine gas in presence of oxygen traces. To prevent the formation of plutonium oxide, addition of $MgCl_2$ is proposed. In this case, the oxides produced by oxygen reduction will be preferentially combined with $Mg(II)$ to produce MgO . MgO is a solid that can be filtered in the reactor core.

The redox potential in chloride media can be controlled, like in fluoride media, by the ratio $[UCl_4]/[UCl_3]$ (for instance, a ratio of 1/100 leads, thermodynamically, to prevent the oxidation of nickel and iron), but chromium is oxidized to $Cr(II)$ in chloride or oxide form. Chromium oxide is known to be highly protective of an alloy exposed to air, which is interesting for shaping phases. In molten salts, the stability of such a layer (of chromium oxide) has to be evaluated. The presence of $MgCl_2$ can decrease the stability of chromium oxide and increases the corrosion rate. These studies are under way at CNRS/IPN Orsay.

5.7.5.2. Component and technology development

This section deals with the pre-conceptual design activities carried out in France on the MSFR concept, and with the development and validation of technology and components related to experimental facilities.

Pre-conceptual design activities on MSR systems. Several simplified designs of the 3 GW(th) MSFR with the Th-U cycle were proposed during the EVOL and SAMOFAR European projects in order to integrate qualitative and quantitative studies related to safety analysis. Safety analysis of the functional design of the fuel circuit was conducted, and the results were used to propose changes to eliminate or reduce the consequences of accidents for each design

option. This continuous process is ongoing to achieve safety-by-design and its advancement depends on the sophistication of the available numerical modelling.

Some illustrations are given here after about the conclusions of the SAMOFAR project on an updated MSFR design; these 2019 conclusions are not publicly available. This design is limited to the functions to be performed by the components and no technical detail is given about them.

For the first time for the MSFR concept, an arrangement of the intermediate salt circuit is proposed to fit a selection of functions and recommendations and to allow a safety analysis. The general approach is the connection of several fuel loops and several loops for power conversion through common collector tanks. Figure 52 shows the functional structure of the intermediate salt circuit between the intermediate heat exchangers (IHX) and the conversion heat exchangers (CHX), the intermediate salt tanks (HT = hot tank, CT = cold tank), and the position of the various elements with respect to the confinement barriers. The pumps are in blue. The startup and shutdown of the circuit is considered as well as the case of the reactor in standby mode (the heat is then removed by a “dumping HX”). The size of the circuit and its fluid volume was calculated to provide damping of the power demand by the conversion that stabilizes the system. The intermediate circuit can be also used to adapt the operating parameters during load following.

FIG. 52. Functional structure of the intermediate salt circuit of the MSFR.

A set of passive, or partly active, decay heat removal devices has been selected and studied. Two emergency draining containers were proposed: An Emergency Draining Tank (relatively frequent draining events are expected because this design allows for reversible draining of the fuel salt), and a Core Catcher (in case of failure of the Emergency Draining Tank). Both are passively cooled by gas through the second (core casing) and third (reactor casing) barriers. This cooling remains to be studied in more detail to assess its reliability because of the possibility of common cause failure. Figure 53 shows a schematic presentation of the intermediate salt circuit (in yellow) in interaction with the fuel circuit (in green; circles are pumps), with several IHX and CHX loops, and with common collector tanks (for maintenance) and accessories. Figure 54 depicts an attempt of plant design in the case of absence of the emergency draining tank. Three confinement barriers are shown in this figure: the fuel casing, the reactor casing, and the reactor building. The decay heat removal system uses natural gas convection inside the reactor casing and the reactor building for two cases: Fuel remaining in the core and fuel drained into the core catcher. Approximate sizes were deduced from simplified calculations.

FIG. 53. Schematic presentation of the intermediate salt circuit of the MSFR.

SWATH facility and associated simulation tools. The SWATH experiments (LPSC, France) developed during the SAMOFAR European project aim at improving and validating numerical

thermohydraulic models of molten salt used for design and safety studies, and more specifically during the draining of the fuel salt. This involves convection (reinforced by turbulence mixing), conduction, and radiation between different phases (solid, liquid and gas) present in some of the MSFR components for various states of the reactor. Due to quite high Prandtl values of the molten salts, the thermal development lengths will be in general larger than the hydraulic development lengths. Accurate predictions of the heat exchanges will thus require a precise understanding of the flow field.

FIG. 54. Attempt of MSFR plant sizing in case of absence of emergency draining tank.

To reduce as much as possible the experimental uncertainties and obtain significant results, detailed definition of the design of the SWATH experiments are in coordination with numerical simulation work. Accordingly, different experimental channel geometries are investigated in both a water model experiment and a salt experiment. This allows performing PIV measurements in the water model experiment to precisely determine the flow conditions inside the experimental channel and thus compare them to the CFD predictions for an isothermal flow [264]. The work detailed in Ref. [265] focused on phenomena that were identified as being a priority for the MSFR concept and the SWATH experiments: The turbulence, the radiative heat transfer modelling, and the solidification of water and of molten salt. A numerical model for simulating the draining process was introduced and used for analysing different draining scenarios [266].

A multi-scale phase change model [267] has been developed to improve the description of the solidification of molten salts and the information available on the properties of the resulting solid phase. Numerical results were obtained using this model, considering natural or forced convection in the liquid phase, to simulate the experimental results obtained with molten salt experiments. A good agreement was observed. Figure 55 shows the temperature and flow fields in the solid and liquids formed during the solidification process. The images on the left and right sides show the solidification experiments without rotation and with rotation, respectively.

FIG. 55. Temperature (bottom) and flow (top) fields during the solidification process [265].

The SWATH facility [266] is operated on a discontinuous working principle in which the flow is established by regulating the pressure difference between two tanks, rather than using a pump. A glove box with an argon atmosphere is included between the tanks to protect a special portion of the circuit that can be changed for testing different configurations. Figure 56 presents a diagram and photo of this facility.

FIG. 56. Schematic (left) and photo (right) of the SWATH facility.

A closed channel section with different shapes or an open channel section can be placed in the glove box. Other types of experiments related to solidification and melting processes have been done to design efficient cold plugs [268], or to validate the solidification model. In the latter case, since it is expected that the flow by convection in the fluid phase has a significant effect on the shape of the solidification front, two different boundary conditions will be investigated: Natural and forced convection. The solidification experiment (see Figure 57) is based on a tube rotating inside an annular cavity filled with molten FLiNaK salt. The rotating tube contains an inner tube to enable the circulation of argon (cooling gas), which decreases the temperature of the external wall of the outer tube below the FLiNaK melting point, to initiate the solidification process. This figure presents the system after extraction from the salt bath with the solidified salt at its bottom; the salt bath is another part of the experiment.

FIG. 57. Picture of the solidification experiment.

FFFER facility. The Forced Fluoride Flow for Experimental Research (FFFER) experiment (LPSC - CNRS Grenoble) started in 2009, and a cold plug design was first implemented, as described below. This facility is a LiF-NaF-KF eutectic (FLiNaK) salt forced convection loop whose main objectives were:

- Study the liquid–gas separation in the case of an on-line physical molten salt processing. Since on-line bubbling is an efficient process to capture dispersed non-soluble particles in a liquid and partly absorb dissolved gas, this process was selected for test with molten salt. It consists in injecting bubbles into the flowing salt in the pipes followed by a liquid–gas separation in another part of the facility.
- Acquire technical experience in operating and designing molten salt experiments at high temperature, including flow control, salt level measurements and safety devices (freeze valve).

A partial description of the facility is given in [268], and Figure 58 offers a view of the facility with thermal isolation around the pipes. Tests of helium bubbling and liquid–gas separation were done in the FFFER facility in 2017, showing satisfactory results in the configuration used (about 1% vol. gas, 1.9 salt litre/s); however, the results have not been published. The design of the separator top has to be improved for running higher flow.

FIG. 58. View of the FFFER facility with thermal isolation around the pipes.

5.8. R&D ACTIVITIES IN ITALY

5.8.1. Activities at Politecnico di Milano

The Nuclear Reactors Group at Politecnico di Milano (POLIMI) (NRG-POLIMI) has been involved in the modelling and analysis of MSR concepts with thermal and fast neutron spectra,

in the context of several international projects (EVOL, SAMOFAR, SAMOSAFER, IAEA's Coordinated Research Project (CRP) T12026 [269]) and collaborations with other research institutions (e.g., TU-Delft, PSI, CNRS-Grenoble, Karlsruhe Institute of Technology, Westinghouse Electric Co., VTT Technical Research Centre of Finland). Simulations and analyses oriented for experiments have been carried out over the last ten years to study different MSR concepts. The need to perform dedicated analyses is due to 1) the scarce experience on these reactor systems compared to the fleet of current nuclear reactors (e.g., Gen II and Gen III, and research reactors), and 2) the uncommon feature of the fuel motion in case of MSRs with liquid fuel that make inadequate the adoption of 'standard' simulation tools for nuclear cores. From a modelling point of view, the main peculiarities that characterize the complex environment of such systems are the drift of the delayed neutron precursors, the strong coupling between neutronics and thermohydraulic behaviours, the internal heat generation in the fluid, and the possibility to perform online reprocessing and online fission product removal. These peculiarities have an impact on the dynamics and control of the system and require specific tools for the simulation of the reactor core [270] and burnup analysis, along with tools for the investigation of the heat transfer characteristics and natural circulation capabilities. The activities of the NRG-POLIMI focused on reactor physics (advanced modelling on multiphysics calculations), system behaviour and operation (dynamics and control), and component and technology development (experimental thermohydraulic activities).

5.8.1.1. System behaviour, operation, safety, and security

The classic control strategies adopted for LWRs and Sodium Fast Reactors are not suitable for MSRs due to the different characteristics of the latter systems. To support the development, testing, and validation of the control system design, simulation tools specifically conceived for control purposes of MSRs are necessary. As a first step, to evaluate the effects of the fuel circulation on the system kinetics, simplified models [271, 272] were preliminarily set up to allow for the decay of delayed neutron precursors in the part of the primary circuit that is out of the core, and they were assessed against the experimental data available from the MSRE (Figure 59, top). The results obtained are relevant for the startup and the shutdown phases of the reactor.

Regarding the analysis of MSR dynamics, different numerical models were conceived for studying the entire system behaviour (core, primary, and secondary circuits), based on lumped and one-dimensional approaches. In this framework, the approximations related to the adoption of simplified geometries were also investigated [273] highlighting the impact of the one-dimensional modelling to predict the reactivity loss outside the active region. These models were adopted for linear stability analyses concerning both thermal and fast MSRs [274]. The outcomes indicate that, especially at low power, the dynamics is faster with respect to the non-circulating fuel system and is dependent on the fuel velocity (Figure 59, bottom). A reduction in the stability margin is possible when the drift of the delayed neutron precursors dominates over the temperature effects, in particular for moderated systems due to the possible positive reactivity effect of the moderator.

FIG. 59. Top: Gain of the transfer function between power variation ($\Delta P/P_0$, where P_0 is the nominal power) and reactivity ($\Delta\rho$) for the MSRE at 8 MW (blue solid line – “Present work”) and comparison with experimental data (red dots – “Experiment”) [272]. Bottom: Inhour

equation for the MSBR with circulating and static fuel (ρ_0 being reactivity compensation for the loss of delayed neutrons and σ being the reciprocal time constant) [274].

A simulator that is more oriented to engineering for the study of the MSFR power plant has been developed in the framework of the SAMOFAR project, based on the Modelica language [275]. The simulator focuses not only on the primary circuit, but also considers the intermediate circuit and the energy conversion circuit. The choice of Modelica allows the use of standard and validated libraries for the modelling of the main components of the power plant, such as pipes, pumps, and turbines. A special effort was made in developing an ad hoc MSR library for the neutronics kinetic model, the flow in one dimension of the salt fuel, and the motion of the delayed neutron precursors. A preliminary version of a decentralized feedback control scheme for the MSFR was proposed, and it was designed using the mentioned tools. Different control schemes were proposed and good results were obtained even if reactivity control by external means is not employed, which can be the case of the MSFR where control rods are not envisaged in the reactor design [276]. The inclusion of the intermediate circuit and the energy conversion circuit in the simulator of the power plant allow also to investigate a more comprehensive control strategy [277] that confirmed the possibility of controlling the MSFR without such means in the mode of full power. In addition, the dynamics, which is fast controlled, for the plant's electrical power confirms the load-following capabilities, while keeping the controlled variables in a safe bandwidth (see Figure 60). The He Brayton cycle is considered as the energy conversion system in this figure.

FIG. 60. Transient response of the control system for the MSFR without employing external reactivity (control strategy of mass flow rates). Left: Mechanical power and gas flow rate. Right: fuel temperatures and fuel mass flow rate.

5.8.1.2. Reactor physics

Analytic, deterministic, and Monte Carlo approaches [278] (Figure 61, top) were developed for calculating the effective delayed neutron fraction, β_{eff} , which is an important reactor kinetics parameter for both safety and control. These approaches were used to consider the inhomogeneous spatial importance of neutron distribution and to obtain an accurate estimation of β_{eff} in the MSFR. With reference to the MSFR, the NRG-POLIMI developed also an extended version of the SERPENT-2 code to study fuel burnup and core material evolution [279] (Figure 61, bottom), and investigated the fuel cycle and the core's physics performance in the long term (in terms of possible initial core loading, actinide burning capabilities, radiotoxicity generation, decay heat, and safety features), which also included a systematic comparison with fast reactors with solid fuel [280]. The latter R&D activity was the subject of the IAEA CRP T12026 "Near Term and Promising Long Term Options for Deployment of Thorium Based Nuclear Energy", in which POLIMI participated.

FIG. 61. Top: β_{eff} for the MSFR: comparison between analytical, based on Monte Carlo (SERPENT) and deterministic approaches (OpenFOAM) as function of the product between the precursor decay constant λ and the fuel salt loop circulation period T [278]. Bottom: Reactivity and uranium insertion rate for the transuranic-started version of the MSFR obtained with the reactivity control algorithm [279].

As a development in the direction of more accurate analyses, a Multi-Physics Modelling (MPM) approach (Refs [270, 281]) was undertaken due to the capability of solving different physics in the same simulation environment. The adoption of this approach allows considering the fluid dynamics (both incompressible and compressible flow with buoyancy effects), the neutronics and the heat transfer in an inherently coupled way. Both steady state and transient analyses (such as reactivity insertion, fuel mass flow rate variations, and chilled inlet) can be performed. In this way, a 2D axial-symmetric model of a core channel of the MSBR, and 2D axial-symmetric [282] (Figure 62, top) and 3D models of the MSFR [181] (Figure 62, bottom) fuel circuit were developed. The multiphysics models were applied using both Finite Element and Finite Volume (FV) solvers, the latter being also open source and with a high parallelization level. In the framework of the SAMOFAR project, the model based on the FV solver (i.e., OpenFOAM) has been improved to study the potential accidents of the MSFR, the impact of the helium bubbling system, and the fuel compressibility effects [283]. To this aim, the neutronics modelling of the multiphysics solver based on OpenFOAM was extended with multi-group neutron diffusion and the simplified P3 (SP3) neutron transport approach [284], and in the thermohydraulic modelling with the inclusion of a compressible thermohydraulic model with two phases, which is based on an Euler-Euler approach [285]. This solver underwent an intense verification phase during the SAMOFAR project, based on a benchmark developed ad hoc for this purpose [286].

FIG. 62. Top: Evolution of power and average core temperature in the MSFR core after a stepwise reactivity insertion equal to 200 pcm (super-prompt-critical) with a 2D axial-symmetric model [282]. Bottom: Velocity and power density fields in the MSFR after an accident of a single pump failure with a 3D model [181].

The fluid dynamics improvement is aimed at studying the effect of the online bubbling system for removing fission products that build up in the molten salt (especially the gaseous and the metallic ones). From a neutronics point of view, a correct bubble spatial distribution is essential to obtain a correct evaluation of the void reactivity feedback coefficient due to the importance of the inherent spatial and neutron effects (Figure 63, left). This solver will be useful in the design of the online bubbling system, in terms of location optimization and in assessing the system's capability for extracting fission products.

To combine the accuracy of the MPM approach with acceptable computing power for the analysis of the overall plant dynamics, a Geometric Multiscale strategy [272] was developed. It consists in modelling the components of the plant according to their dimensionality. In particular, a 3D core channel of the MSRE was modelled with an MPM approach, whereas the

components located out of the core were modelled with a zero-dimensional approach (Figure 63, right).

FIG. 63. Left: Bubble distribution (void fraction) in the MSFR obtained with the two-phase compressible solver [285]. Right: Conceptual scheme of the Geometric Multiscale model of the MSRE [272]; DNP is Decay Neutron Precursors, and MP is Multi-Physics.

The thermohydraulic behaviour of internally heated molten salts flowing in straight circular channels was analysed using analytic and numerical tools (Refs [270, 287]), with application to MSRs moderated by graphite and to the MSFR (see Figure 64, top). In particular, it was shown that the effect of internal heat generation can affect the wall-fluid bulk heat transfer characteristics and that it can be described by means of a corrective factor to be applied to traditional correlations (e.g., the Dittus-Boelter or Gnielinski correlations) for the Nusselt number. The developed correlation was used to carry out a parametric investigation of the effect of decay heat on the components that are out of the core of the MSFR. The volumetric power causes higher temperatures at the channel wall than in the channel's inner parts, but the effect is significant only in case of large diameters, or low velocities, or both. This effect can be of interest in the case of the channels of the salt reprocessing system [288, 289]. As for the dynamics of natural circulation in a molten salt with internal heat generation, linear analysis methods were developed to investigate the natural circulation features of internally heated fluids for MSR applications [290, 291], along with an advanced approach based on CFD for studying spatial effects [292]. These methods allow studying the stability of the operational conditions in terms of both asymptotic behaviour (modal analysis; see Figure 64, bottom) and initial transient energy growth (non-modal analysis). They were compared with non-linear models including a system code approach (one-dimensional discretization with use of pressure drop and heat transfer correlations) and an approach based on CFD [293].

FIG. 64. Top: Comparison among heat transfer correlations and CFD results obtained by means of FLUENT software, in the Reynolds range of interest for MSRs ($Pr = 11$) [270]. Bottom: Stability map (modal analysis) of a natural circulation loop for various levels of internal heat generation, from $\alpha = 1$ (conventional, localized heat flux only) to $\alpha = 0$ (internal heat generation only) [293].

5.8.1.3. Component and technology development

A testing facility (DYNASTY) was built at POLIMI [294] (Figure 65, left) aimed at 1) the experimental investigation of the natural circulation dynamics in presence of distributed heat generation, with a specific focus on the analysis of thermohydraulic instabilities (stability maps); and 2) the validation of the theoretical models and simulation tools developed to study the natural circulation capability of a loop with a homogeneously heated molten salt, useful for the MSFR design optimization. DYNASTY is a natural circulation loop that can operate in circulation that is driven by buoyancy or with forced flow since it is equipped with a pump and

electric heating elements. The latter allow the possibility of creating different conditions in terms of natural circulation configurations (conventional and distributed cases). The facility can run with a solar molten salt (KNO_3 , NaNO_2 , NaNO_3), but also can be operated with water and water–glycol mixtures, the latter being a simulant of the thermohydraulic behaviour of a molten salt. The facility has been extended (Figure 65, right) within the SAMOFAR project to also investigate the dynamics of two coupled loops with natural circulation.

As mentioned, the facility has been characterized in detail through a strong modelling effort using various modelling approaches. This includes the development of models based on a system code approach, CFD models based on the Large Eddy Simulation approach that can unveil flow reversal features during unstable transients, and data assimilation techniques.

FIG. 65. Left and centre: The DYNASTY facility built at POLIMI Labs. Right: in orange, the extension of the facility (e-DYNASTY) to study coupled natural circulation loops.

5.8.2. Activities at Politecnico di Torino

The faculty staff at Politecnico di Torino, Energy Department, belonging to the NEMO (Nuclear Engineering MOdelling) group have been involved in research activities on the neutronics of MSRs since more than two decades ago, in the frame of European projects, IAEA CRPs, and collaboration with researchers at the international level.

The research activities were initially focused on the study of the neutronics of reactors with circulating fuel, aiming at providing proper physico-mathematical models for its analysis. The study of dynamics of circulating fuel was performed at first focusing on the neutronics part, then moving to a multiphysics framework, including the coupling to thermohydraulic phenomena. In more recent years, starting with the European project SAMOFAR, the different expertise available within the NEMO group has been exploited to carry out research activities with a stronger focus on safety.

A brief synthesis of these various activities is provided in the following paragraphs for these two main lines of research, with reference to publications for further reading.

5.8.2.1. Reactor physics: neutronics of circulating fuel systems

The former Reactor Physics Research Group at Politecnico di Torino was involved at the beginning of the 1990s in the European Project IABAT (Impact of the Accelerator Based Technologies) on the safety of nuclear fission. One task was devoted within this very broad and ambitious project to study a design of a subcritical system with liquid, circulating fuel. The neutronics analysis of this kind of systems posed questions regarding the correct way to describe the behaviour of the neutron population within the reactor, especially regarding the movement of delayed neutron precursors that are dragged by the fuel motion before emitting delayed neutrons. Taking profit of the available expertise in the typical methods for reactor physics (e.g. reactor dynamics [295]), various approaches to the neutronics analysis of this kind of systems were addressed, such as the application of perturbation theory [296] and the definition of a point model suitable for the description of the peculiar physical effect appearing when the fissile material is moving [297, 298]. In particular, the issue related to the reduction

of the role of delayed neutrons and the consequent impact on the dynamic response of the system was already addressed in these early works [299].

The involvement on other European projects, fully focused on the MSR concept, allowed to further proceed in the neutronics analysis of systems with circulating fuel. The MOST project (Review of molten salt reactor technology, 2001-2004) gathered various European institutions interested in analysing the state of advancement of the MSR concept. In this framework, the staff at Politecnico di Torino carried out a research activity on the development of kinetic models for MSRs, leading to a consistent definition of the factorization-projection process adopted in point kinetics and quasi-statics [300, 301]. This process was then applied to transients of interest for this kind of systems, such as modifying the fuel flow with consequent oscillation of the reactivity and, thus, the power (see Figure 66).

FIG. 66. Top: Power evolution, normalized to initial value, for different numbers K of shape recalculations for a transient of fluid velocity in a critical system (ΔT_ϕ is the time interval adopted for the shape recalculation). Bottom: Time behaviour of α (kinetic parameter of point kinetics related to reactivity evolution, see detail in Ref. [301]), normalized to its initial value α_0 , for different numbers of shape recalculations (solid line: $K = 1$; dashed line: $K = 2$; circles: $K = 10$; dots: $K = 50$; diamonds: $K = 100$; plus sign: $K = 1000$).

During the MOST project, various institutions developed computational tools for the simulation of MSRs as a function of time, with specific focus on the MSRE, which was carried out in the 1960s at ORNL. The performance of these tools was compared in a benchmark exercise with experimental measurements obtained on the MSRE (see Figure 67).

FIG. 67. Pump start-up transient in MSRE (JEF decay data) (left) and Natural convection transient in MSRE (right) [302].

The MOST project was followed by the ALISIA project (2007-2008), in which the activities at Politecnico di Torino on MSR dynamics were further pursued. In particular, the aspects related to the influence of the fluid-dynamic aspects on the neutronic behaviour were studied, with an approach initially based on single-physics simulations: the fuel velocity field was modified parametrically to evidence its neutronic effects [303], and the effect of periodic oscillations of reactivity was analysed [304]. The neutronic and thermohydraulic tight coupling in an MSR then required a more complete simulation of MSR dynamics, leading to a series of activities aiming at the development of a model for the coupled neutronic-thermohydraulic analysis of an MSR, thanks to the collaboration with researchers from LANL in the implementation of suitable numerical schemes for the solution of the coupled system of equations [305, 306, 307, 308].

The MSR activities were continued in the frame of another European project, EVOL (2010–2013), where Politecnico di Torino was involved in another benchmark focused on the design

of the MSFR. The definition of a reactor design with a fast spectrum, with a profound difference in terms of geometry of the core and fluid patterns, was the motivation for the activities carried out, focusing on the neutronic behaviour of the reactor in steady state, and relevant parameters for safety analysis and potential breeding of new fissile material, as the possibility of a fuel cycle based on thorium was also foreseen [309]. The result of the comparison among the project partners was summarized in Ref. [233].

In parallel to these European projects, the IAEA launched a CRP on “Studies of advanced reactor technology options for effective incineration of radioactive wastes”, where a task was devoted to the analysis of molten salt systems for the purpose of their application in the framework of Partitioning and Transmutation (P&T) of nuclear waste. The scientific community involved was constituted by the same participants of the previous European projects, and the results of such joint effort are summarized in Ref. [310].

The activities carried out in the frame of these variety of projects also stimulated the interest in more fundamental aspects of the physico-mathematical modelling of these topics, leading to the collaboration with other researchers on these topics. The mathematical characteristics of the neutronics problem for a system with fluid fuel has been studied, comparing neutron transport and its diffusive approximation [311, 312], while the problem of reactivity oscillations due to localized modification of the core composition was approached by the logic of assessing its statistical properties [313].

More recently, in the frame of the SAMOFAR project (2015-2019), an Uncertainty Quantification (UQ) study concerning the influence of the uncertainty in the nuclear data on main neutronic parameters was carried out. Using both a well-established technique like Generalized Perturbation Theory (GPT) and a reduced order approach known as XGPT, both available in the Monte Carlo code Serpent-2, the uncertainty on the effective multiplication parameter was estimated for the main nuclides of interest, that is, ^{232}Th and ^{233}U . To accomplish this task, the nuclear data generation and processing procedure has been used, through the application of codes such as NJOY, SANDY and the T6 package, to generate perturbed cross section libraries as input for Serpent. The UQ study proved that the uncertainty related to the aforementioned nuclides seriously impact the final uncertainty on the multiplication parameter, suggesting to extend the study also to macroscopic cross sections, employed in deterministic multiphysics calculations, in order to propagate the uncertainty also to the main thermohydraulic parameters [314].

5.8.2.2. *Safety assessment and uncertainty quantification for the MSFR*

Politecnico di Torino enlarged the field of research activities pertaining to MSRs in the SAMOFAR project, where a new activity focused on risk analysis and safety assessments was launched. It participated in the project as part of the CIRTEN Consortium (Consorzio Interuniversitario per la Ricerca Tecnologica Nucleare - Intra-University Consortium for Nuclear Technology Research), together with Politecnico di Milano, and the contribution of various faculty of the NEMO group allowed to approach this new field of research with specific attention on the safety characteristics of the MSFR design.

The Integrated Safety Assessment Methodology (ISAM) proposed by the GIF's Risk and Safety Working Group (RSWG) in 2011 was selected as the basic methodology [315], and the ISAM was reviewed to better reflect the international standards and rules, and to suit the peculiar case of the MSFR. The approach described in the IEC EN 61508 constitutes a

milestone for design that is safety driven in the process industry and represents an inspiration for this analysis. Its major idea is that the systems safety must be studied and pursued from the early design by risk analysis tools through the definition of Safety Instrumented Functions to be analysed in order to understand the effective risk reduction needed in terms of safety systems and additional safety requirements.

The idea applied was to enrich the ISAM with the following risk analysis tools in order to select a list of hazards as complete as possible and improve the efficiency of the analysis and the detailed design definition: Functional Failure Modes and Effects Analysis (FFMEA), Master Logic Diagram (MLD), and Lines of Defence (LOD). This work was summarized in the SAMOFAR deliverable D1.5.

The implementation of the methodology started with the identification of deviations that can compromise system safety (in terms of Postulated Initiating Events (PIEs), the most challenging conditions for plant safety), through two approaches applied at the same time: the FFMEA, a bottom-up approach, focused on the identification of the functions of the system and the analysis of the consequences of the loss of each of them, and the MLD, a top-down approach, that after the selection of a top event identifies its possible elementary causes. In addition to the identification of PIEs, the FFMEA and the MLD allowed to highlight the lack of information on some systems, procedures or phenomena, to point out potential limitations of the design, and to make suggestions to enhance the safety of the concept [248]. The LOD method was then applied for selected accident scenarios to ensure that every accident evolution of the reactor was always prevented by a minimum set of homogenous (in number and quality) safety features before a situation with potentially unacceptable consequences might arise. Each event was briefly characterized, identifying also plausible prevention measures. During the application of the LOD method, some input data regarding natural behaviour of the plant following the initiating events, with a preliminary evaluation of expected radiological consequences, were fundamental to be able to define the number of safety provisions. The LOD helped to identify that additional provisions could be necessary to ensure the complete management of the accident (e.g. the addition of a core catcher or equivalent), or to recognize the importance of ensuring the availability of some existing components.

The results of this analysis were included in the SAMOFAR deliverable D1.6, and presented in Refs [316], [317], and the functional approach successfully applied to the fuel circuit in SAMOFAR has been chosen for the analysis of the fuel treatment unit in the SAMOSAFER project (2019-2023), aiming at contributing to the development of the design of this unit with an approach that is risk oriented.

Within the SAMOSAFER project, Politecnico di Torino is also contributing to the analysis of the reactor system with the development of methods and models for the early detection of deviations from the operational status of the systems, in order to reduce the need for action of safety provisions.

5.9. R&D ACTIVITIES IN JAPAN

As described in Section 2.1, US Development Efforts, ORNL achieved various accomplishments for the MSR development roughly from the 1950s to 1970s. Based on their results, Kazuo Furukawa and his group in Japan have been investigating the MSR “FUJI”, which is moderated by graphite, and related accelerator technology since the 1980s [318, 319].

FUJI is mostly based on the MSBR and DMSR designs with some improvements. His group has provided comprehensive design results, such as detailed design results of FUJI-U3 achieving a 1.0 conversion ratio (self-sustaining) by using ^{233}U -Th fuel salt. The FUJI concept is described in Appendix V (Graphite Moderated MSR), among other MSR concepts. In addition, they investigated the following areas: the FUJI-Pu using Pu as start-up fissile material, transmutation capability of minor actinides, the super-FUJI of 1,000 MW(e), and a mini-FUJI as a pilot plant.

As for the initial fissile feeding for an MSR, Furukawa proposed a concept of Accelerator Molten-Salt Breeder (AMSB) in 1981 with a numerical study [320]. AMSB is similar to an Accelerator Driven System (ADS), which utilizes a proton accelerator and molten salt target containing thorium to produce a large amount of ^{233}U for feeding fissile material to an MSR. Inspired by A.A. Harms's book [321], Furukawa combined an MSR and the AMSB as the Thorium Molten Salt Nuclear Energy Synergetic System (THORIMS-NES) in 1990 [322], and a dry-reprocessing facility for molten salt fuel is placed between these two facilities, as shown in Figure 68.

FIG. 68. Thorium Molten Salt Nuclear Energy Synergetic System (THORIMS-NES).

In order to promote MSR development, Furukawa and his group established the International Thorium Molten-Salt Forum (ITMSF) in 2008, and the Thorium Tech Solution Inc. (TTS) in 2010. ITMSF is an observer member of the GIF-MSR System Steering Committee from this Committee's beginning.

Under the Atomic Energy Society of Japan, a special committee was started in 2013 as "Nuclear Application of Molten Salt." Within this committee, a working group with ITMSF members proposed general design criteria (GDC) for an MSR [319], which is based on the draft GDC from the ANS201 committee, where the GDC for FHR was discussed in the US.

Also, a comprehensive summary of MSR safety analysis is provided by ITMSF, where the philosophy for the accident analysis of MSRs with safety criteria and 40 possible accidents with some numerical results are described [319].

Besides the above activities, there are basic studies in the 2010s related to MSRs at Japanese universities and research institutes, such as the University of Fukui, the University of Tokyo, the Central Research Institute of Electric Power Industry, and the National Institute for Fusion Science.

The Japanese government started to support the development of MSR technology from 2019. Three MSR venture companies are selected; two are promoting an MSR with fluoride salt that is moderated by graphite, and one is promoting a fast MSR with chloride salt.

5.10. R&D ACTIVITIES IN THE NETHERLANDS

5.10.1. Activities at NRG

5.10.1.1. Introduction

As the Dutch national nuclear laboratory, NRG operates the High Flux Reactor (HFR) and the related Hot Cell Labs (HCL) in Petten and carries out a nuclear R&D program sponsored by the Ministry of Economic Affairs and Climate. An important part of this R&D program is the study of innovative nuclear systems, and in particular MSR. The overall aim of the Dutch molten salt program is to gain experience with the handling, irradiation, post-irradiation research and waste treatment of molten salts, as well as the modelling of a reactor in which the fuel is liquid. The goals for the program for the short and medium term are to:

- support qualification of materials and components for use in an MSR;
- improve insight into the behaviour of fission products in molten salt in relation to accident scenarios and decommissioning;
- develop a technique for the (partial) on-line removal of fission products (“helium bubbling”);
- test and qualify measurement and control techniques required to operate MSRs;
- design, build and operate a facility for the HFR that can serve as a prototype for future first-of-a-kind reactors (a so-called molten salt loop).

The program has matured in the past few years, and currently consists of several irradiation, modelling and laboratory efforts. Important collaborations are those with key partner JRC, which provides in-depth knowledge of salt thermochemistry and thermophysical properties, with the Delft University of Technology, where Ph.D. students work on salt chemistry and helium bubbling, and Research Centre Rez. Part of the program is carried out in the European Horizon 2020 project SAMOSAFER, coordinated by TU Delft, which started in September 2019. JRC Karlsruhe has provided and continues to provide well-characterized fuel salt samples for the salt irradiations carried out in the HFR Petten, and Research Centre Rez has provided the salt samples needed for gamma irradiation tests.

5.10.1.2. Reactor physics

In order to allow simulation of systems fuelled with molten salt, NRG extended the SPECTRA code [323, 324, 325], which is a thermohydraulic system code developed at NRG, designed for thermohydraulic analysis of NPPs. The code was originally developed for LWRs, but the flexible code set-up also allows application to High Temperature Reactors (HTRs), Liquid Metal Fast Reactors (LMFRs), and with the current extension also to MSRs. The code can be used for accident scenarios involving loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in NPPs. Models include multidimensional two-phase flow, non-equilibrium thermo-dynamics, transient heat conduction in solid structures, and a general package for heat and mass transfer with built-in models for steam/water/non-condensable gases, including natural and forced convection, condensation, and boiling. For molten salt and liquid metal reactor applications, the fluid properties and heat transfer correlations can flexibly be defined by the user. A point reactor kinetics model is available, with an isotope transformation model to compute concentrations of important isotopes (for example, ^{135}Xe). The package for radioactive particle transport in the code deals with

radioactive fission product chains, release of fission products, aerosol transport, deposition, and resuspension.

With all extensions made to the SPECTRA code, the code can model the following relevant phenomena for a system fuelled with molten salt:

- delayed neutron precursor drift;
- fission product transport in molten salt reactors, for example, ^{135}Xe ;
- noble gas and noble metal behaviour (based and validated on Refs [72, 326, 327])
- noble metal extraction (data comparison to Ref. [328])
- chromium leaching and deposition (based on Ref. [329])

Ref. [330] offers a more elaborate description of the code and its application to the MSRE. An example output of the code is shown in Figure 69. The figure shows the noble gas ^{135}Xe distribution in the MSRE at steady state.

FIG. 69. Steady state distribution of noble gases in MSRE using the SPECTRA code.

5.10.1.3. Fuel and coolant chemistry and supporting technology

Work at NRG is focused on irradiation and modelling efforts, while JRC Karlsruhe and CV Rez have provided the fuel salt samples for the salt irradiations in the HFR (see Figure 70). Work on supporting technologies is therefore generally carried out in the context of the irradiation projects. Supporting work currently consists of gamma irradiation tests, laboratory testing to optimize helium bubbling conducted at the TU Delft, and electrochemical measurements on fluoride salts in support of the SALIENT-03 (SALt Irradiation Experiment) project.

SAGA. During preparations for the SALIENT-01 fuel salt irradiation tests, it became clear that radiolytic fluorine gas production as observed and studied previously at ORNL [331, 225, 226] is an issue affecting the quality of irradiation experiments in the HFR. A capsule irradiation in the HFR runs for many cycles, each of 30-31 days, with intermediate shutdown periods of 4-30 days. During these shutdown periods, samples cool down quickly to below 65°C. It was observed at ORNL that at temperatures below 150°C, the rate of radiation-induced bond cleavage in MSRE salt exceeds the rate of bond re-formation through thermal motion, resulting (with varying delays) in the production of fluorine gas. The corrosive gas F_2 is itself a manageable safety hazard, but perhaps more important to the quality of the experiment is that this leaves the salt sample in a reduced (and non-representative) state. It is not yet clear whether the same issue occurs for chloride salts.

FIG. 70. Overview over the HFR.

An experiment with the name of SAGA (SAlT GAMMA) was designed with the objective of measuring the efficiency and amount of F₂ gas released from salt samples in a gamma field, and to study the mechanism of radiolytic gas production for several salt species. The idea of the experiment is to continuously measure pressure in sealed salt-filled capsules during high-dose gamma irradiation using HFR spent fuel (at an average gamma dose rate of ~30 kGy/hour in air-filled ionization chambers). Absorption of radiation is registered by ionization chambers and converted to sample dose using a Monte Carlo N-Particle (MCNP) model, and temperature in the system is monitored by thermocouples (45-60°C). The SAGA irradiation facility is reloadable. For the first experiments, it has been loaded with powder samples of LiF, BeF₂, UF₄, ThF₄ and a LiF-BeF₂-UF₄ mixture, with one empty reference capsule. A collaboration with the Nuclear Research Institute Rez in the Czech Republic was established for the fabrication of the fuel salts and fluorination of the capsules.

Construction of the facility and loading with salt samples was finalised in the last quarter of 2019 (see Figure 71). The first results of the gamma irradiation were recorded in early 2020. The observed pressure build-up at least confirms the expected trend that heavier (higher-Z) salts absorb more energy and therefore generate more fluorine gas. In 2020 the fluorine gas will be recombined with the salt samples by heating the system to above 100°C, after which the irradiation experiment will be repeated to check for consistency of results.

FIG. 71. Left: one of the salt-filled SAGA capsules before sealing. Right: the completed assembly of 6 capsules before loading into the SAGA facility.

Helium Bubbling. In a collaboration between NRG and TU Delft on chemistry of molten salts, research has been conducted with focus on the chemical speciation of the main fission products in fluoride salts, and on the on-line noble metal extraction via helium bubbling. A CFD model for 2-phase flow has been used by NRG to model flotation of noble metals; this model is to be validated by comparison with laboratory tests on the flotation efficiency. Preliminary CFD calculations have been carried out by NRG to establish the influence of helium bubbles on the temperature profile under irradiation (see Figure 72). The simulations show that bubbling promotes heat transport to the edges of the salt column, resulting in a more uniform temperature distribution, but also that the influence of bubbling on the temperature profile is small except for high flow rate and small bubbles. Figure 72 gives results for the smallest bubble size tested of 0.2 mm and flow rates ranging from 5 to 50 µl/s. Results for larger bubbles which are not shown here, reveal that the temperature profiles with and without bubbles are practically the same. However, the temperature profiles are clearly different in the case of bubbles with a size of 0.2 mm and a flow rate of 50 µl/s. The simulations therefore confirm that the new SALIENT test can be conservatively designed based on ‘no bubbling’ conditions.

FIG. 72. Left: salt velocity profile. Center and right: axial temperature profiles for different flow rates of bubbles of diameter 0.2 mm (Center: profile at the centre of the salt column; right: profile at the salt-capsule interface).

5.10.1.4. Performance of Materials

ENICKMA. Nickel-based alloys are foreseen to be used in MSR systems for their high corrosion resistance. Nickel is however sensitive to thermal neutron irradiation, producing helium which consequently may cause embrittlement of the material. The ENICKMA irradiation project is being prepared to study this embrittlement in nickel-based materials. For this scoping study, the irradiation targets have been defined and material suppliers have been contacted. Table 15 presents an overview of the various materials. 316 L(N) is a 316 steel with low carbon content and added nitrogen developed by CEA in the framework of the ASTRID sodium-cooled fast reactor. GH3535 is a Chinese analogue of the well-known nickel-molybdenum-chromium alloy Hastelloy N, MONICR (MOlybdenum-Nickel-CRromium) is a closely related alloy developed in the Czech Republic, and Hastelloy 242 has a relatively high molybdenum content for better high-temperature strength. Figure 73 shows MONICR samples as obtained from Research Centre Rez (CV Rez, Czech Republic).

The effect of helium embrittlement will be investigated by different techniques. After irradiation, tensile and low cycle fatigue samples will be loaded to study irradiation effects. Small punch test samples will also be loaded to study the plastic deformation behaviour and foils for Transmission Electron Microscopy (TEM) are included to investigate the microstructural behaviour after irradiation. Concept design was completed in 2018; in 2019 safety assessments were conducted with nuclear and thermomechanical analysis. Start of irradiation is currently foreseen for early 2021.

TABLE 15. OVERVIEW OF THE VARIOUS MATERIALS FOR THE ENICKMA IRRADIATION EXPERIMENT

Material	Supplier
316 L(N)	CEA (Fr)
Hastelloy N	Haynes (US)
GH3535	SINAP (Ch)
HN80MTY	COMTES FHT (Cz)
MONICR	COMTES FHT (Cz)
Hastelloy 242	Haynes (US)

FIG. 73. Left: A segment of MONICR plate as supplied by CV Rez. Right: tensile samples fabricated at CV Rez from the same material.

SALIENT-03. Following SALIENT-01 and considering the lessons learned from that experiment (see below), SALIENT-03 is a molten salt capsule irradiation focused on the in-pile corrosion of Alloy N. This experiment is performed in collaboration with JRC, and is described in Subsection 5.6.3.

5.10.1.5. System behaviour, operation, safety and security

Transient simulations for molten salt systems have been performed using the SPECTRA code described before. For a system cooled by molten salt (Mk1 PB-FHR), simulation results of a loss of primary coolant flow and natural convection cooling through a passive Direct Reactor Auxiliary Cooling System (DRACS) loop were compared to results obtained using the US RELAP5 and SAM codes [332]. Figure 74 presents a snapshot of the simulation. Ref. [333] shows that the results with SPECTRA are in good agreement with the other two US codes. Differences found are mainly related to modelling assumptions for the pumps.

FIG. 74. Transient results at a particular time for Mk1 PB-FHR using the SPECTRA code.

Simulations for the Thermal-Convection Heat Removal (TCHR) test of the MSRE were performed recently by Alcaro [334]. This required a precursor simulation to model the state at the start of the transient representing the ‘history’ of the system prior to the start of the transient. Based on available data from existing ORNL reports dating back to the MSRE era, this was implemented in the SPECTRA code. Simulation results of this transient showed good agreement with measured data from the MSRE.

5.10.1.6. Component and technology development

SALIENT-01. SALIENT-01 (see Figure 75 and Ref. [223]) is the first molten salt fuel irradiation conducted since the 1960s (when salt capsule irradiations, salt loops and the MSRE were operated at ORNL). The primary goal for the irradiation was to obtain hands-on experience. Graphite was selected as the crucible material because molten salt corrosion is not considered an issue for this material. The salt composition, $78\text{LiF}\text{-}22\text{ThF}_4$ (99.9% ^7Li , natural ^{232}Th), was selected based on the possibilities for synthesis and purification at JRC Karlsruhe in Germany at the start of the project in 2015 (these possibilities have significantly expanded since then). LiF-ThF₄ is also the basis for the European MSFR concept.

The irradiation rig consists of a stack of 4 samples of $\sim 1.5\text{ cm}^3$ of $78\text{LiF}\text{-}22\text{ThF}_4$ salt in open graphite crucibles, allowing passage of fission gases into the 1st containment space. A 5th graphite crucible at the top of the stack is added as a reference. The five crucibles are enclosed in a sample holder made of 316 steel providing double containment. The 1st (inner) containment is sealed by welding.

In the design phase, it was found that the fuel samples were susceptible to radiolysis upon cooling to below 150°C, whereby fluorine gas is released [225, 226]. These concerns prompted the start of the SAGA gamma irradiation project (Subsection 5.10.1.3), and were the reason for the termination of the twin experiment SALIENT-02. Following an evaluation of the

consequences of radiolytic F_2 production, the SALIENT-01 irradiation was started on the 10th of August 2017 and finished on the 17th of August 2019, after 17 cycles (508 Full Power Days) of irradiation.

Specific questions to be addressed were:

- How far does the fuel salt penetrate into the graphite, and what does fission product penetration into the graphite look like?
- Is fission gas release indeed ~100%, as expected based on the low solubility of Xe and Kr in molten salts?
- What is the ultimate size distribution of noble metal fission product particles observed on nickel foil and on graphite, and what is the relative deposition rate ('sticking factor')?
- Does prolonged contact with salt have an influence on the surface quality of the graphite?
- Can effects of radiolytic fluorine gas production (uranium carbide formation, interstitial fluorine in the graphite, and CF_4 gas in the plenum space) be observed?

These questions should be answered by an on-going post-irradiation examination campaign in the Petten Hot Cell Laboratories. Planned examinations include gamma spectrometry, puncturing of the 1st containment for fission gas analysis and electron microscopy with chemical analysis.

FIG. 75. Left: Loading of the LiF-ThF₄ salt samples in an inert glovebox environment at JRC Karlsruhe. Right: The SALIENT-01 sample holder after irradiation. Following cutting of the instrument lines in the DM-cell of the HFR, the sample holder is inserted into a gastight holder for transport to the Hot Cell Laboratories.

5.10.2. Activities at TU Delft

5.10.2.1. Introduction

Delft University of Technology (TU Delft) has a long tradition with homogeneous nuclear reactors, starting with the vision of Jan Went, the first professor of nuclear reactor physics at TU Delft and director at the KEMA research centre, to develop a homogeneous reactor in Europe. In 1957, the KEMA opened a new nuclear laboratory for the development of the KEMA Suspension Test Reactor (KSTR), a reactor with small uranium-thorium-oxide fuel particles in a water flow pumped through the primary circuit. By continuously refuelling particles and cleaning the coolant, uninterrupted operation over a long period was envisaged. After construction and studies for ten years in a zero power reactor, the construction of a reactor with thermal power of 1 MW started in 1963. After operation for a few years, this reactor was shut down in 1977. Meanwhile, Prof. D.G.H. Latzko worked in the 1960s on the use of molten salts as heat transfer fluids for power generation.

The interest in homogeneous reactors revived at TU Delft at the beginning of this century as part of a research program on Generation-IV reactors. It was concluded that breeding of new fuel and complete destruction of long-lived actinides could best be accomplished in a homogeneous reactor. Research on the thermal Molten Salt Reactor started in 2005 and focused on the thorough understanding of the Molten Salt Reactor Experiment (MSRE), and on

improving the thermal Molten Salt Breeder Reactor (MSBR) design, both originally developed by ORNL. Various alternatives were investigated, including one-fluid, one-and-a-half fluid and two-fluid reactor core designs. In addition, the chemical and physical properties of various fuel salts were studied, in close collaboration with the JRC-Karlsruhe. Although the thermal MSR design clearly has a lot of potential, research at TU Delft now focuses on the Molten Salt Fast Reactor (MSFR) design originally proposed by CNRS and adopted as the reference design for Generation-IV in Europe. This culminated in two projects funded by the European Community (EC): SAMOFAR, running from 2015-2019, and SAMOSAFER, which started in 2019.

TU Delft coordinated the EU SAMOFAR project and is currently coordinating the follow up project SAMOSAFER. It contributes with fundamental research regarding the computation challenges in the MSFR (coupled neutronics-CFD, melting and solidification of salts), numerical and experimental work to measure viscosity and other parameters of the (fuel) salt, as well as various applications like assessing safety by analysing transients and the design of freeze plug devices and passive mechanisms for decay heat removal.

Furthermore, TU Delft investigates the phase equilibria and thermodynamic properties of several types of fluoride and chloride salt systems (fuel relevant compositions with inclusion of fission products), their chemical interaction with structural materials, and the chemical speciation of fission products generated during irradiation using a combination of experimental studies and thermodynamic modelling assessments using the CALPHAD method (see Subsection 5.6.1.1).

Numerical and experimental work on extraction of fission products via gas bubbling and part of the thermodynamic modelling studies is carried in the framework of a scientific cooperation between TU Delft and NRG.

Many of the results below are described in the PhD theses of Refs [335, 336, 337, 338, 339, 340, 341, 342].

5.10.2.2. *Numerical code development and transient analysis*

During the SAMOFAR project, completed in 2019, the TU Delft multi-physics simulation package, originally based on the neutron diffusion code DALTON [343], has been re-developed and extended combining a state-of-the-art discrete ordinates neutron transport code (PHANTOM-SN) with a newly developed simulation code for fluid dynamics (DG-Flows). This multi-physics code package [344] was verified with code to code comparisons with other SAMOFAR partners and separate benchmarks from the literature [345]. The package was utilized to investigate the behaviour of the MSFR under both steady state and transient conditions. The transients studied were Unprotected Loss Of Heat Sink (ULOHS), Unprotected Loss Of Fuel Flow (ULOFF), Total Loss Of Power (TLOP), Unprotected Pump Over-Speed (UPOS), and salt Over-Cooling (OVC). The main conclusion from these simulations is that the reactor response to these scenarios is excellent. No major weak points were identified regarding the MSFR design's safety: modest temperature variations, and power variations are strongly damped by negative reactivity feedback. An example of the fuel salt's temperature distribution obtained by multi-physics analysis is shown in Figure 76 below.

FIG. 76. Evolution of the fuel salt's temperature distribution during a TLOP transient [342].

5.10.2.3. Development of reduced order model

Multi-physics analyses of the three-dimensional MSFR are inherently expensive from a computational point of view. A strong effort was made for developing Reduced Order Modelling (ROM) to partially counteract this cost at TU Delft. The focus has been on methods that are non-intrusive to be able to work with complex models without requiring access to the underlying model equation, i.e. a black box approach [346]. The newly developed methods are based on adaptive sampling of the parameter space combined with interpolation for the amplitudes of a set of modes that adequately describe the system, the latter determined by proper orthogonal decomposition (POD). The set of modes and their amplitudes are adaptively enriched by sampling the parameter space where the errors are largest until specified error criteria are met. The methodology was also extended to transient scenarios where time is treated as any other input parameter, which enables efficient handling of parametric dynamic models. The reduced order model was subsequently used for various purposes, such as uncertainty quantification [347]. Steady state analysis has been performed for 30 independent input parameters originating from cross sections, neutron precursors and thermodynamics parameters. The various parameters' influence on the observables mentioned, like k-eff and the maximum temperature, has been ranked and probability distributions of various key parameters, such as k-eff and the maximum temperature, were made. An example of such distributions is shown in Figure 77 below.

FIG. 77. Probability distributions of the maximum salt temperature in the core and of the multiplication factor, k-eff, under influence of input parameter variation.

5.10.2.4. Generalized polynomial chaos development and application

TU Delft has also made significant progress in developing and applying methods of Generalized Polynomial Chaos (GPC) expansion to perform fast and accurate uncertainty quantification and sensitivity analysis of the MSFR despite the computationally expensive multi-physics modelling required. Just like ROM, GPC also represents a black box approach that builds efficient meta-models of selected important system responses as a function of uncertain physical/chemical properties and undefined design parameters. It has the advantage of using multi-dimensional polynomial approximations (instead of hierarchical linear interpolation as in the ROM approach); therefore, it can sometimes be more efficient in terms of overall computational cost than ROM methods, especially in cases where strong non-linearity is present and only key performance parameters are of interest [348, 349]. Moreover, it provides a straightforward approach to derive local sensitivities as well as sensitivity metrics based on global variance (Sobol indices). GPC was used to perform a detailed analysis of the MSFR steady state (using high-fidelity coupled neutron transport–CFD calculations). Figure 78 shows the results for the salt temperature in the reactor ($1/16^{\text{th}}$ of the rotationally symmetric design), highlighting that a) the upper parts of the core have a non-negligible probability of surpassing the maximum temperature design specification of 1023°K , and b) that the lower parts of the heat exchanger and the vessel have a similarly non-negligible probability of cooling

below the minimum temperature specification of 923°K. Thus, GPC allows the type of in-depth analysis that can identify parameters with significant effect on performance metrics, thus helping to optimize the MSFR.

FIG. 78. Left: Domain where $T < 1023^{\circ}\text{K}$ with probability higher (red) and lower (blue) than 95%. Right: Domain where $T > 923^{\circ}\text{K}$ with probability higher (red) and lower (blue) than 95%.

GPC was also used to quantify the uncertainty regarding the salt composition. As the MSFR will use molten salt as fuel, knowing the physicochemical properties of the salt is paramount under all conditions. This is heavily hindered by the lack of data available and the inherent difficulty of predicting properties for such multicomponent systems, containing many different elements due to the emergence of fission products during operation. TU Delft has made significant first steps in addressing this issue by combining expertise and GPC tools with the CALPHAD method (see Subsection 5.10.2.12), allowing quantifying the uncertainties in the phase diagrams and thermodynamic properties computed using the optimized thermodynamic models (see Subsection 5.10.2.12). Figure 79 shows a result of such a calculation, where the uncertainties on Gibbs energy parameters were propagated to the mixing enthalpy values of a LiF-KF system, and compared with experimentally determined values, showing good agreement. The true power of such a coupled GPC-CALPHAD methodology — that is, using CALPHAD (for predicting phase diagrams and other properties) in combination with GPC (providing an efficient connection between the input and output parameters of CALPHAD) — is that by analysing the relation between the uncertain input parameters and the experimentally determined uncertainties, the input parameters can be tuned such that the resulting phase diagrams are in accordance with experiments, and these values can be used to also give statistical information about parameters that can be calculated, but are difficult to measure.

FIG. 79. Comparison of measured and CALPHAD simulated mixing enthalpy values together with derived uncertainties. Red, green, and blue lines represent the 0th, 50th and 100th percentiles of the calculated phase diagram, while the black circles showcase the measured values together with their experimental uncertainties. The black dotted line corresponds to the mixing enthalpy computed using the thermodynamic model implemented in the JRCMSD database (see Subsection 5.10.2.12).

5.10.2.5. Ultrasonic technique for measuring viscosity of radioactive fluids at high temperature

An innovative method based on ultrasonic waves propagation is being developed for the simultaneous determination of the viscosity and density of the molten salt fuel at elevated temperatures. A thin plate is used as a waveguide to transmit shear waves separating the transducer from harsh environments such as elevated temperatures, highly corrosive fluids and/or radioactive fluids. At the solid–fluid interface, the echo signal of the ultrasonic wave depends on the operating frequency, the physical properties of the fluid (viscosity and density), and the properties of the wave guide (density and shear modulus). The research at TU Delft

deals with the study of the factors for determining the salt fuel's physical properties, and the solutions required to maximise the accuracy of the proposed technique, especially at very low viscosities. The rheology of non-Newtonian fluids is under investigation as well.

5.10.2.6. *Physicochemical effect of mixing molten salt fuel and water*

The physical effects of mixing were studied for the hypothetical situation in which molten fuel salt and water come in contact. A series of scoping experiments were performed to understand the interaction of solid fuel salt, which was used at the MSFR, with water in various concentrations to simulate different situations such as fuel leakage in water or vice-versa. The analysis included investigating the dissolution in water of fuel salt (LiF-ThF₄ or LiF-ThF₄-UF₄) and FLiNaK (LiF-NaF-KF 46.5-11.5-42 mol%). The measurements were performed under both irradiating (gamma source) and non-irradiating conditions to understand the effect of irradiation on the salt solubility. The solvent was examined by chemical analysis after the dissolution experiments for evaluating the solubility of the cations and the formation of hydrate compounds. Ref. [339] provides additional information on the physicochemical effect of mixing molten salt fuel and water.

5.10.2.7. *Development of a fast, passive freeze plug*

Passive protection is provided by a so-called freeze plug in the MSFR. The freeze plug is an actively cooled blockage consisting of solidified (frozen) salt. In the case of a postulated accident, such as a station blackout, the plug will melt because the active cooling will stop, enabling the reactor vessel contents to flow into underground tanks. This process must be completed within 8 minutes at most to prevent damage to the reactor vessel and components in the MSFR. One of the promising designs consists of a thick distribution plate that contains holes of a certain diameter; see Figure 80. This design increases the melting surface and diminishes the drainage time of the reactor vessel. Plate material and geometry were varied. Based on preliminary calculations, the drainage time can be reduced to 6 minutes, which is below the prescribed 8 minutes. The use of additional cooling fins is under investigation.

FIG. 80. Preliminary design of a freeze plug for the MSFR. Fins are installed to enhance the melting of the plugs that are present in the distribution plate.

5.10.2.8. *Molten salt-structural materials interaction*

The suitability of the structural material is defined to a large extent by its resistance against corrosion by the molten salt. The MSRE experience showed that Hastelloy-N, a Ni-Cr-Mo alloy, could withstand the harsh operational conditions in the reactor (high temperatures, contact with the corrosive salt, high radiation dose), and prospective alloys in current MSR designs include the chemical elements in this alloy. The rate of corrosion is primarily determined by the redox potential of the salt, which is controlled by the UF₄/UF₃ ratio when a molten salt based on fluoride is used. During irradiation with this kind of salt, free fluorine is formed which reacts with UF₃, hence increasing this ratio, and leading to oxidation reactions such as $\text{Cr}(\text{alloy}) + 2\text{UF}_4(\text{salt}) \rightarrow \text{CrF}_2(\text{salt}) + 2\text{UF}_3(\text{salt})$, and subsequent formation of voids in the Ni-based structural alloy. Among the large variety of MSR designs under investigation, reactors with and without redox control or salt clean-up are under consideration. Because Ni is

the main element of the structural alloy, and Cr is the most likely to be dissolved, the phase equilibria of the fuel salt with potential Ni and Cr corrosion products has been recently studied. A thermodynamic model for the AF-NiF₂, AF-CrF₃ (A = Li, Na, K), and CrF₂-CrF₃ systems was developed for the first time. The model can feed simulation codes that model the behaviour of the liquid fuel during normal and accident conditions.

5.10.2.9. *Chemical speciation of fission products*

The importance of chemical speciation of fission products in nuclear fuels is beyond doubt. The fate of these elements and their influence on the fuel properties strongly depend on their chemical state, which in turn depends on the reactor parameters such as temperature and redox potential. For reactors that utilize molten fluoride salt mixtures as nuclear fuel, a thorough thermochemical analysis of the most important fission products is not yet complete. In order to predict the stable phases and the physicochemical properties of the mixtures, thermochemical analysis and studies of phase diagrams are employed. Two thermodynamic databases are under development at TU Delft, one including the main salt-soluble fission products (i.e. caesium and iodine), and the other including the noble metals and their fluoride phases.

The chemistry of caesium and iodine is particularly relevant as their potential release into the environment is a subject of primary concern for the safety of the population and the environment. The isotopes ¹³⁵Cs, ¹³⁷Cs, and ¹³¹I are indeed the main cause for the radiological consequences of an accident and their behaviour in nuclear fuels must be carefully evaluated. A comprehensive thermodynamic assessment of the Li, Cs, Th/F, I system was therefore performed at TU Delft in collaboration with JRC-Karlsruhe and NRG, and combines experimental investigations of phase diagrams, vapour pressure measurements, and thermodynamic modelling.

For the fission products consisting of noble metals, Mo, Ru and Nb are particularly relevant. Mo and Ru have the highest fission yield and are expected to be in the metallic state under the reducing conditions which would normally be maintained in the reactor as a result of the redox control of the salt during operation. The behaviour of Nb (dissolved in the salt or precipitating) depends directly on the redox potential conditions. Knowledge of their thermochemistry, and in particular of the formation of noble metal fluoride phases, is essential to understand their fluorination behaviour and establish a proper fission product management strategy. A thermodynamic model of the noble metal elements Mo-Nb-Ru-F has been developed for this purpose, by combining calculations based on first principles and modelling assessments of the same type as CALPHAD.

5.10.2.10. *The influence of melting and solidification on heat transfer*

A good understanding of the melting and solidification phenomena of the LiF-ThF₄-UF₄ fuel in the MSFR is required for the design of the freeze plug, a key safety component, for an accurate prediction of the possible formation and growth of a solid layer of salt on the reactor walls during normal operation and for the analysis of accident scenarios where solidification of the fuel salt might pose a risk. As such, the goal of this research, being part of the SAMOSAFER project, is to improve the knowledge and understanding of the physics underlying the process of phase changes. To this end, the effects of phase change will be included in a computational fluid dynamics (CFD) model based on the Discontinuous Galerkin (DG) approach through the so-called enthalpy method, where the melting/solidification front is tracked implicitly. Validation of the applied numerical models will be performed through a

set of experiments of phase changes, for both laminar and turbulent flow regimes. In addition, the role of forced and mixed convection on the process of these changes will be investigated. Both types of convection may occur in an MSFR during normal operation.

5.10.2.11. *Noble particles extraction*

One of the advantages of having a liquid fuel is the possibility of controlling and extracting fission products while operating the reactor. At present, helium bubbling is seen as the most mature technique for in-core removal of fission products, and it can be employed to extract the gaseous fission products. Moreover, this process has also the potential to remove via flotation the insoluble particles, such as noble metals, that might otherwise deposit on relatively cold metallic surfaces. A feasibility and performance study for the removal of particles via flotation is ongoing at TU Delft in collaboration with NRG. It aims to gain a better understanding of the underlying mechanisms involved in the process, and evaluate the dependence of the extraction efficiency with the process parameters. An experimental setup using simulant fluids was developed for this purpose to provide the required data (local, instantaneous particle concentrations) for process optimization and scale-up for molten salt systems. Measurements of the particle concentrations with the help of Laser Induced Fluorescence are on-going for particles with sizes varying from about 100 nm to micrometers.

5.10.2.12. *Development of a large thermodynamic database for multi-component systems*

A large thermodynamic database, known as JRCMSD, has been developed over the years by the JRC-Karlsruhe (see the section on European Commissions' contribution) for fluoride and chloride salts. It is based on the CALPHAD methodology and quasi-chemical formalism in the quadruplet approximation that allows the computation of thermodynamic equilibrium properties of multi-component systems. The models in this database rely on the great wealth of phase diagram and crystallographic studies produced by the original MSRE program at ORNL, as well as other miscellaneous sources, and more recent measurements performed at the JRC. Since 5 years ago, TU Delft has contributed to the development of this database with experimental structural and thermodynamic studies, and modelling activities, in a joint effort with JRC. Fission product (Cs, I, Nb, Mo, Ru) systems, and corrosion product (Ni, Cr) systems, as mentioned previously, were investigated as part of this effort. Moreover, the LiF-UF₄, NaF-ThF₄, and KF-ThF₄ systems were recently re-examined, as part of an effort to systematically review and model the AF-AnF₄ systems containing actinides (A = Li, Na, K, Rb, Cs; An = Th, U).

5.10.2.13. *Structural studies and advanced structural-thermodynamic modelling*

It is known that depending on conditions of composition, temperature, and redox potential, the local structure of the molten salt (an ionic liquid) can widely vary with cases where the ions in the melt are completely dissociated, form molecular species, or even exhibit some degree of network formation (polymerization). The local structure of the melt can be directly related to the thermodynamic and transport (viscosity, thermal conductivity) properties. A fundamental understanding of the relationship between the local structure of the molten salt and its physicochemical and thermodynamic properties is needed to gain greater predictive capability over the dynamic (far from ideal) behaviour of the fuel salt. Through in-situ Extended X-ray Absorption Fine Structure (EXAFS) spectroscopy studies at high temperature in the molten states, interpreted with the aid of Molecular Dynamics, the structural characteristics of the AF-AnF₄ melts (A = Li, Na, K, Cs; An = Th, U) were explored. A dedicated furnace set-up for

molten salts was developed at the TU Delft for the EXAFS measurements of hygroscopic, corrosive, and radioactive salts. Moreover, in combination with the quasi-chemical formalism, CALPHAD models that reproduce the structure of the melt simultaneously with the phase diagram and key thermodynamic properties (such as mixing enthalpies, heat capacities, and activity coefficients) were developed and optimized. This is ongoing work, but to date, the advanced models were successfully applied to LiF-BeF₂, LiF-AnF₄ (An = U, Th), NaF-ThF₄, and AF-ZrF₄ (A = Na, K).

5.11. R&D ACTIVITIES IN THE RUSSIAN FEDERATION

Over the last twenty years, R&D on MSR in the Russian Federation has been mainly focused on fast spectrum concepts with or without thorium support [122, 350, 351]. These concepts were recognized by GIF as an alternative for the long term to fast neutron reactors with solid fuel, and with attractive features, namely strong negative reactivity feedback coefficients, smaller fissile inventory, and simplified fuel cycle. A negative coolant temperature reactivity coefficient is universally recognized as a desirable safety feature for power reactors. For example, the Molten Salt Actinide Recycling Transmuter (MOSART) concept can operate within technical limits using an alloy with high nickel content as container material, and with different fuel loadings and make-up based on TRUs (from used LWR fuel with MA/TRU ratio up to 0.45) as special actinide transmuter, as self-sustainable system (conversion ratio (CR) = 1), or even as a breeder (CR > 1).

Basic information is provided concerning the fuel cycle of the MOSART power plant [351] integrated with facilities of the Experimental Demonstration Centre (EDC) (see Figure 81, where the blue oval indicates the back end of the nuclear fuel cycle). The EDC is under construction in Russia at the site of the Mining and Chemical Combine (MCC), and after 2020 will begin reprocessing of spent nuclear fuel (SNF) from VVER-1000 reactors on the basis of innovative technology, providing recovered nuclear material (refined products) for recycling in thermal and fast reactors with solid fuel. The highly active raffinate, containing long-lived minor actinides, is sent for conditioning, as shown by the light blue arrow in this figure. Using MOSART, most of the uranium and plutonium can be recycled to be used as solid fuel for thermal and fast reactors (dotted green line in the figure), and this process may reduce the volume and radiotoxicity of radioactive waste. The main feature of the MSR technology in this application is the flexibility for using pyroprocessing of spent fuel that was cooled for a short time and multiple recycling for better use of resources, reduction of waste, and gaining additional profits as compared with the traditional fuel cycle using solid fuels.

It was proposed to use the technological capabilities of the MCC site to place a MOSART system in the immediate vicinity of an aqueous reprocessing plant for SNF, linking it to the EDC infrastructure. The main design objective of this 2400 MW(th) MOSART with a single fluid and no fertile material is to close the nuclear fuel cycle for all actinides, including neptunium, plutonium, americium and curium. It is assumed that the fuel cycle of this complex will be organized as follows (see Figure 81): the bulk of the removed uranium and plutonium returns to thermal and fast reactors with solid fuel, and the remaining transuranic elements are transferred for utilization in the MOSART system. The co-location of the MOSART and the SNF reprocessing plant will provide the MCC site and the surrounding customers with electricity (7.92 TW(e)hrs per year) and reduces the problems of transport of nuclear materials and radwaste management. As with fluid fuel based on fluorides, the entire fabrication process of solid fuel elements is avoided, providing for exceptional flexibility. The fuel can be directly

mixed with the reactor's primary system as needed at any time. In addition, there is no need for long cool times and interim storage. Thus, a significant part of the front-end effort (including radioactive doses for workers) and cost is eliminated for MOSART. All fresh fuel of fluoride molten salts containing significant quantities of fissile materials for initial loading and make-up will be manufactured onsite at the EDC by the hydrofluorination process.

FIG. 81. Nuclear fuel cycle with MOSART at MCC site (ILW means intermediate level waste).

The main advantages of MOSART are the ability to widely vary the minor actinides content in the fuel salt without losing the inherent safety, and the absence of fabrication of solid fuel and re-fabrication. Hence, there are significant proliferation resistance and safeguards implications related to the fuel make-up and chemical processing in a MOSART plant: (1) there will be continuous variation of isotopic concentrations in the fuel salt from both actinides transmutation and chemical processing; (2) the refuelling scheme include the ability to continuously feed the core with fresh fissile material; and (3) the plate-out of noble metals in the primary circuit could complicate inventory tracking. In addition, the fuel becomes less attractive for fissile material diversion after each recycling. During 50 years of operation, a 2,400 MW(th) MOSART can utilize more than 12 t of minor actinides.

To reduce the possibility of diversion of nuclear material, the MOSART plant is integrated (1) at the front end with the VVER SNF aqueous reprocessing plant, and (2) at the back end with the high temperature fuel salt clean-up facility; all facilities are located at the MCC site.. In a pyroprocessing facility for molten salt, the higher actinides would always accompany the plutonium, so this operation would never produce a "clean" material that would be attractive for diversion.

The Russian Federation established an experimental complex in support of future R&D on MOSART. The test reactor facility consisting of a 10 MW(th) MSR (LiF-BeF₂-AnF_n fuel salt) with a homogeneous core at the MCC site is currently under design within the framework of a project supported by State Corporation ROSATOM. Major achievements of recent MOSART R&D are summarized below.

5.11.1. ISTC Project #1606 (2001-2008)

Various Russian institutes, including Zababakhin Institute of Technical Physics (Snezinsk), KI (Moscow), Institute of Chemical Technology (Moscow) and Institute of High Temperature Electrochemistry (Ekaterinburg), started the ISTC # 1606 project "Experimental study of molten salt technology for safe and low waste treatment of plutonium and minor actinides in accelerator driven and critical systems" in 2001. These institutes already made contributions to MSR technology, including key technical solutions for a promising MOSART concept with single stream. Experimental and theoretical studies were performed in the following main areas: (1) neutronics and thermal hydraulics of the reactor unit, (2) key physical and chemical properties of fuel salt, and (3) container materials for the fuel circuit. The main results achieved in this project are summarized next.

The 2,400 MW(th) MOSART system developed within this project has a homogeneous core with intermediate-to-fast spectrum of neutrons. The fuel salt is a molten 15LiF-58NaF-27BeF₂ (mole%) mixture with the addition of plutonium and minor actinides fluorides. The specific

power of the fuel salt is about 43 W/cm^3 . To minimize actinide losses in reprocessing, a removal time of about 300 days for soluble fission products (rare-earth trifluorides) is considered. Lithium is enriched to 99.99% ^7Li . There is no strong requirement on gas permeability ($10^{-8} \text{ cm}^2/\text{s}$) for the graphite reflector (0.2 m) of the MOSART core, but molten salt should be excluded from the open pore volume (pore structure $< 10^{-6} \text{ m}$). The 2,400 MW(th) MOSART core can satisfy the most important neutronic and thermal-hydraulic considerations: (1) $\text{AnF}_3 + \text{LnF}_3$ concentration in the fuel salt is truly within the solubility limit of $\text{AnF}_3 + \text{LnF}_3$ for molten $15\text{LiF}-58\text{NaF}-27\text{BeF}_2$ (mole %) at a minimum fuel salt temperature in the primary circuit of 600°C for the fuel cycle scenarios under consideration; (2) regions of reverse, stagnant or laminar flow are avoided, and (3) the maximum temperature of solid reflectors is low enough to allow their use for the suitable time of 4 yrs.

Several nuclear data libraries, codes, and computation models were employed to compute safety-related neutronics parameters for the 2,400 MW(th) MOSART system. The results show that the parameters are favourable for reactor safety, mainly due to the strong density and fuel Doppler effect. A preliminary study of transients has demonstrated that the MOSART design is an inherently stable reactor design on account of its large, negative fuel temperature coefficient in combination with its negative graphite reflector reactivity coefficient. The MOSART reactor is not expected to be seriously challenged by the major, unprotected transients such as loss of flow without scram, ultimate loss of heat sink, and overcooling. A preliminary evaluation indicates that the MOSART system has attractive features of performance and efficient transmutation of actinides, while having lower amounts of total materials and of waste than prior MSR designs, including the MSBR.

Experimental studies for molten LiBeNa/F system, was found quite wide range with minimal of LiF (17-15 mole%) and of BeF_2 (27-25mole%) content in the ternary composition, which provide fuel salt able to get PuF_3 solubility of 2 and 3 mole%, respectively, at 600°C , and to keep adequate melting point ($< 500^\circ\text{C}$). In the design of MOSART, the material that is specified for nearly all metal surfaces contacting the fuel and coolant salts is an alloy (HN80M-VI, HN80MTY), which is a Russian modification of commercial Hastelloy N. The results of a loop corrosion experiment during 1,600 hours with on-line redox potential measurement demonstrated that operations at high temperature with a LiBeNa/F salt are feasible using carefully purified molten salts and loop internals. In an established interval of salt redox potential of 1.25-1.33 V relative to a Be reference electrode, corrosion is characterized by uniform loss of material from the surface of samples with very low rate of corrosion. No significant change in the corrosion behaviour of samples of HN80M-VI and HN80MTY alloys due to the additions of plutonium trifluoride and tellurium to the LiBeNa/F salt.

5.11.2. ISTC Project #3749 (2008- 2016)

The purpose of the ISTC# 3749 project consisted in developing the Th-U MOSART concept, mainly in improving the fuel circuit configuration and system safety and in experimental studies on the basic properties of fuel salt and the fuel salt's compatibility with structural materials.

The main goal of this project consists in the selection and experimental study of properties of molten salt compositions and metallic structural materials for prospective nuclear power systems and fuel cycle facilities. Eutectic mixtures, containing lithium, calcium and thorium trifluorides were selected for physical-chemical studies. The following characteristics were

experimentally obtained for selected compositions: melting temperature and enthalpy, temperature dependences of density, isobar thermal capacity, viscosity, thermal conductivity, and plutonium trifluoride solubility. The electrochemical properties of two metals (neodymium, plutonium) and their fluorides, dissolved in the selected molten salts, were investigated. Processes of reductive extraction of neodymium, lanthanum and thorium trifluorides from molten fluoride salts into liquid bismuth, as well as a process of re-extraction of neodymium, lanthanum and thorium from a system consisting of liquid bismuth/molten lithium chloride, were studied experimentally. The compatibility of fuel salt containing tellurium additions with nickel-based alloys, at controlled redox-potential and different mechanical loadings on the alloy samples, was tested. Tests on compatibility of nickel-based alloys with the selected melt were carried out in a thermal convection loop under controlled redox-potential. A detailed investigation into the change of the structure and strength properties of the alloys after about 1,000 hours of exposure in circulating molten salt was completed. Investigations on hydrogen permeability of molten fuel salts and nickel alloys also were carried out.

The concept of a Th-U MSR with homogeneous core and fast neutron spectrum was developed. Several scenarios for fuel loading of initial and feeding compositions of transuranium elements from the spent fuel of LWRs as well as thorium, as a fertile material, were considered. It was found that an MSR with a homogeneous core offers flexibility in the fuel cycle, as the MSR plant is able to operate with a wide spectrum of fuel and fertile material loadings without reactor shutdown and no need of changing the core design. Such plant may quickly be re-oriented and included in any strategy of a nuclear power program.

Therefore, a competitive development of an MSR with a homogeneous core both as Th-U breeder and as a new element in a nuclear power program for burning of transuranium elements from spent nuclear fuel is possible. Thermal hydraulic calculations were carried out for the most promising variants of this system. Safety and non-proliferation characteristics also were considered.

5.12. R&D ACTIVITIES IN SWITZERLAND

5.12.1. Motivation

The MSR R&D activities in Switzerland were launched in 2013 and concentrate within the ETH domain (domain of the Swiss Federal Institutes of Technology). They are carried out predominantly at Paul Scherrer Institut (PSI), and at the universities ETH Zurich and EPFL Lausanne through students' projects. Historically, there was a dedicated MSR research project already in the 1970s and 1980s at the PSI predecessor, EIR (Eidgenössisches Institut für Reaktorforschung), which is described in the history section. The work focused on fast chlorides MSRs and the last concept developed was entitled SOFT [37] (Salt reactor On site reprocessing Fast converter Task). A reactor with 3GWt power and 4 loops layout was proposed, which was based on natural chlorides without isotopic enrichment (see Figure 82).

FIG. 82. Layout of the SOFT reactor containment [37].

Switzerland is nowadays a member of the GIF, and in November 2015 signed a memorandum of understanding and joined the GIF MSR project. PSI acts as the Swiss implementing agent in GIF. The PSI Nuclear Energy and Safety Division (NES) is also a national competence centrum for nuclear reactors. Together with the two major topics, safety of current reactors and waste management, the advanced reactor research belongs to the NES missions. The motivation for advanced reactors research, and especially for MSR, is driven by:

1. The potential of these reactors to combine unique safety features with high fuel utilization and waste minimization.
2. This research offers many novel and multidisciplinary topics.
3. This research is attractive for students and has potential for funding from the granting agencies like the Swiss National Science Foundation (SNSF).

The research activity on innovative MSR technology also represents an important asset to maintain excellence in nuclear engineering, to ensure successful recruiting of highly motivated workforce in the coming years, and to provide an attractive working area for future generations of scientists and engineers. At the same time, the unparalleled combination of high fuel utilization with unique safety features of advanced reactors can promote a shift of public acceptance towards nuclear energy in the future. MSRs have potential for both, risk reduction and radioactive waste minimization. Moreover, once the technology is mastered and advanced, the inherent safety and possible simplicity of MSRs may allow for capital cost reduction.

The MSR research at PSI integrates several past and ongoing national and international projects. The most important is the participation in the European Horizon 2020 projects SAMOFAR and SAMOSAFER. The national projects relate predominantly to sustainability and safety. The SNSF financed in 2014 a PhD project dedicated to designing a modular MSR for low waste production. The major outcome of this PhD project was the proposal of breed-and-burn fuel cycle [131, 128, 132] in MCFR [352]. Another project entitled “Feasibility and plausibility of innovative reactor concepts in an European electricity supply environment” was financed in 2015 by ETH Zurich and the Fund for Projects and Studies of the Swiss Electric Utilities. Finally, Swiss Nuclear, the nuclear energy section of Swiss Electric Utilities, financed in 2016 a project focusing on “Chemical thermodynamic aspects of LWR Pu and MA burning in MSR.”

The MSR research in Switzerland is based predominantly on students’ work. Several MSc and PhD students already defended their thesis at PSI or in cooperation with PSI. The research at PSI focuses on MSR safety and sustainability. It, however, covers a wide range of different topics, which can be structured into four research areas (see Figure 83). The assessment of MSR safety is the ultimate objective. The other three areas address the MSR sustainability, the fuel salt chemistry for nominal and accidental conditions, and the transient analysis and decay heat removal (DHR). Knowledge from the last three areas is necessary to evaluate the MSR safety.

FIG. 83. Illustration of four research areas and their interdependencies.

5.12.2. Major achievements

In three of the four research areas mentioned above, the focus is set to modifying existing tools or developing dedicated tools, so that the MSR peculiarities can be fully addressed by the simulation. The major developed or modified codes are:

- EQL0D & EQL3D for equilibrium fuel cycle procedures;
- GEMS (Gibbs Energy Minimization Software) for thermodynamics modelling;
- AMoDy for molecular dynamics;
- GeN-Foam, a multi-physics tool for MSR core analysis;
- TRACE-PARCS system code for MSR transient analysis.

From the above listed codes, TRACE-PARCS was not developed at PSI and was only modified for MSR applicability. EQL0D and EQL3D are procedures dedicated for analysis of the fuel cycle at equilibrium. EQL3D was originally developed for simulation of fast reactors with solid fuel, and later modified for MSR application. EQL0D was developed from the beginning as a dedicated MSR tool. Also the multi-physics solver GeN-Foam, based on the OpenFOAM, was originally developed for reactors with solid fuel. Nowadays, it includes several necessary models for MSR simulation and some additional models are being implemented. GEMS is being developed at PSI since the year 2000 for several different purposes. In the past, it was applied for pyrochemical reprocessing of molten salt, hence some of the necessary compounds are already included in its thermodynamics database, which nonetheless needs further extension and modification for MSR application. The AMoDy is a molecular dynamics tool being developed at PSI with some unique features, as described below.

5.12.3. Fuel cycle simulation (EQL0D)

An MSR with liquid fuel can be designed as a breeder reactor in both U-Pu and Th-U cycles, and it can also burn the legacy waste from spent fuel or utilize enriched uranium. In all cases, the equilibrium fuel composition will determine the neutronic performance. Therefore, the Equilibrium procedures EQL0D and EQL3D were developed at PSI. The EQL3D procedure [353, 354] is based on ERANOS [355] and include modifications [280] to simulate the continuous removal of fission products in MSRs. Later, another 0D procedure named EQL0D, specifically dedicated to MSR simulation, was developed [356, 357, 358]. The latest version (v3) of the procedure couples MATLAB with the Monte Carlo code SERPENT [188] through the burn-up matrix and solves the Bateman equation by the same CRAM method [359] as SERPENT. The previous versions, v1 and v2, are MATLAB scripts coupled through the reaction rates of actinides with the cell transport code ECCO [355] or SERPENT, respectively. Accordingly, they did not account for fission products.

Three basic MSR designs were compared using the EQL0D v2 without fission products model: graphite moderated MSR, homogeneous fluoride fast MSR, homogeneous chloride fast MSR. For each of these three options, there were two different fuel salts considered. The six resulting options were compared with the performance from 10 reactors with solid fuel [360].

EQL0D v3, including the fission products model, was applied for a scoping study of graphite and non-graphite moderated MSRs. Five different fluoride salts and six different moderators were evaluated. For each of these 30 combinations, the fuel composition at closed Th-U cycle equilibrium was simulated for many salt-to-moderator ratios and salt channel diameters. Figure 4 presents the results of thousands of EQL0D simulations. For better readability, the reactivity

is plotted only for values above one. The values on the right side of each square in Figure 4 corresponds to performance of the respective salt in an infinite homogeneous fluoride fast MSR without moderator.

Based on Figure 4, the LiF salt is the best performing carrier salt in thermal spectrum being followed by the LiF-BeF₂ salt. Moderators based on beryllium and deuterium are the best and outperform graphite. These two types of moderator would however need coating or cladding, which was not accounted for in Figure 4. Accordingly, the resulting core performance may differ. showed that only SiC as coating or cladding can preserve this performance.

EQL0D v3, was also applied to homogeneous fluoride and chloride fast MSRs. Five potential fluoride and three potential chloride carrier salts have been analysed in the closed Th-U and U-Pu cycle. Figure 84 shows the breakdown of the reactivity excess. Ref. [360] gives a description of the breakdown method. The highest reactivity excess, and so the best performance in both fuel cycles, is provided by the Na³⁷Cl and Ac³⁷Cl₄ carrier salts. Their tremendous reactivity excess in the U-Pu cycle can even enable the utilization of a breed-and-burn fuel strategy [128]. The best performance from the fluoride carrier salts has ⁷LiF, the reference carrier salt of the EU Horizon 2020 SAMOFAR and SAMOSAFER projects. Hence, this salt was used for several fuel cycle studies. A parametric spectral study at equilibrium cycle was accomplished with the EQL3D procedure [354]. A hybrid spectrum MSR core, with thermal and fast spectrum zones was evaluated [165], the simplified reprocessing scheme assessed [361], and the continuous and batch-wise reprocessing schemes compared [362].

FIG. 84. Comparison of eight fast MSR designs using excess and hypothetical (for zero parasitic captures) reactivities with Th-U cycle (left) and U-Pu cycle (right).

The EQL0D was also applied to breed-and-burn fuel cycle analysis since 2015 [128, 131, 132]. At the beginning of the analysis, graphite moderated and homogeneous fast MSRs were considered for both fluoride and chloride salts and in both Th-U and U-Pu fuel cycles. Accordingly, all possible combinations were analysed (eight combinations from 2×2×2): moderated or unmoderated, chloride or fluoride salt, Th-U or U-Pu fuel cycle. As expected, the graphite moderated MSRs have insufficient performance for this type of fuel cycle in both Th-U and U-Pu cases and for both fluoride and chloride salts (dashed lines in Figure 85). Nonetheless, the obtained minimal subcriticality was surprisingly good for the fluoride salt and the Th-U cycle (red dashed line in top Figure 85). The homogeneous fluoride fast MSR showed bad performance and cannot be used for the breed-and-burn fuel cycle (see full lines in top Figure 85). The homogeneous chloride fast MSR performs well in the U-Pu cycle (blue full lines in bottom Figure 85). The reactivity is on the edge and the breed-and-burn cycle is not possible in the Th-U fuel cycle (see red full line in bottom Figure 85). Since the U-Pu cycle provides sufficient excess reactivity, several additional cases were considered with higher actinides share in the salt or with a mixed fuel cycle of U-Pu and Th-U. For all chloride salt cases, the enriched salt with ³⁷Cl was applied. The mixed Th-U and U-Pu cycle seems to be possible (purple full line in bottom Figure 85).

Figure 85 shows the simulation results for an infinite lattice. The major difficulty of designing a breed-and-burn core is the minimization of neutron leakage. The chloride salts are transparent

for neutrons and the leakage minimization results in bulky cores, which may not be acceptable [131, 132, 135]. One possible option can be the multi-zone or multi-liquid core layout [134].

FIG. 85. Evolution of infinite multiplication factor in a breed-and-burn reactor as a function of fuel discharge burnup [132].

5.12.4. Thermodynamic simulation (GEMS)

The Gibbs Energy Minimization Software package (GEMS)¹⁵ is a PSI in-house open-source thermodynamic modelling package that plays a vital role in MSR research. It has been developed at NES since 2000 and was and is applied in numerous EU and international projects like ACSEPT, SACSESS, SAMOFAR, SAMOSAFER and TCOFF, and in a number of national projects financed by consortium of Swiss utilities Swiss Nuclear: “Chemical thermodynamic aspects of LWR Pu and MA burning in MSR”, ATHESC, ASTAM and MOCHATF. This package is applied for thermodynamic modelling in different areas, like LWR related systems (such as crud formation, and water chemistry), waste management, modelling the release of fission products during normal and accident conditions and, with the extended HERACLES database [363] to MSR related systems (simulation of phase diagrams and speciation). GEMS has a modular structure and flexible database with interface built around an efficient open-source GEMS3K numerical kernel code [364] for solving phase equilibria in complex non-ideal systems. Moreover, the extent of GEMS applicability can be significantly broadened by coupling this code with other transport- or multi-physics codes. GEMS3K also contains the TSolMod code library [365], which has more than 25 state-of-the-art mixing models for aqueous, gaseous, solid solutions, and melts. This gives GEMS a broad applicability to various types of materials, such as molten salts, if extended with the necessary datasets, and mixing models for binary interactions between the compounds.

For applications to nuclear materials and reprocessing of spent fuel from LWRs, the HERACLES [363] thermodynamic data base (TDB) for low pressure and high temperature non-aqueous systems has been developed and maintained at PSI since 2010. The database contains thermodynamic properties of pure compounds and binary interaction parameters for various systems. Currently HERACLES database covers data for over 650 condensed compounds, including melts and liquid condensates, and over 350 gaseous compounds, and is valid up to at least 3000°K, covering the majority of the elements of interest.

The HERACLES database is used to study the behaviour of complex systems, speciation in selected molten salts, and Gibbs energies of formation of different species in selected melts. Figure 86 compares vapour pressures for different species of the LiF–ThF₄–CsF mixture calculated using the HERACLES database with experimental and calculated results from literature [221]. Figure 87 shows the vapour pressure as a time evolutions during salt heat up for a more complex LiF–ThF₄–UF₄–Cs–I system evaluated with GEMS.

¹⁵ Kulik, D.A., Dmytrieva, S.V., Wagner, T., Kosakowski, G., Thoenen, T., et al., Gibbs Energy Minimization Software (GEMS) (2014), <http://gems.web.psi.ch>

FIG. 86. Vapor pressure data for the $\text{LiF-ThF}_4\text{-CsF}$ system calculated at PSI with GEMS and compared Ref. [221].

FIG. 87. Vapor pressures for the $\text{LiF-ThF}_4\text{-UF}_4\text{-Cs-I}$ system calculated with GEMS.

Further on, this approach has been used to assess the standard potentials for Mo or MgO species in the LiCl-KCl and LiF-AlF_3 melts to build the E-pO^{2-} stability diagrams and to provide broad and thorough picture of their speciation [366].

By coupling GEMS with the severe accident code MELCOR [367], the cGEMS code was created [368]. It was applied on a generic pool filled by MSFR fuel salt with dissolved I and Cs fission products. The salt was slowly heated up by the decay heat. Figure 88 shows the evaporation behaviour during this heat up process. The cGEMS code can be used for a more comprehensive accident analysis of MSR once the detailed description of the reactor confinement and accident sequences will be available and more fission product elements have been added to the database.

FIG. 88. Fission product release from molten salt pool during the heating process.

5.12.5. Molecular dynamics simulation (AMoDy)

The PSI in-house Molecular Dynamics (MD) tool, AMoDy,¹⁶ is another important asset for MSR analysis. In cases when the information on binary interactions between different compounds in binary systems is missing, the so-called atomistic modelling is used to fill the information gap. In particular, the MD approach is used to study mixing enthalpies, excess volumes for different binary systems, polar as well as non-polar [369, 370] structures, thermal conductivity [371, 372], and diffusion in liquid as well as in solid state. As an example, the information on the excess properties can be evaluated (see Figure 89). In this case, AMoDy allows for direct estimation of mixing properties and serves as a reliable tool in predicting these properties for mixing models in thermodynamic modelling, and at the same time providing information at the atomistic level, thus improving the understanding of the underlying processes.

¹⁶ Nichenko, S. AMoDy [Source code]. <https://bitbucket.org/sergiinichenko/amody>

FIG. 89. Excess density (left) and excess Gibbs energy (right) of the Li-K-F system at 1,100°K. Left: The dots show the calculated values, and the line is a linear interpolation between two consecutive dots.

AMoDy also can be potentially used for the direct estimation of phase diagrams for the systems of interest. One such modelling experiment is presented in Figure 90, where the phase diagram for LiF-KF is shown together with six snapshots of MD results. The snapshots refer to 24% (left side) and 75% (right side) of LiF share in the mixture for three different temperatures. Unstructured regions on the top left and right snapshots represent molten region, the structured zones on the middle and bottom snapshots represent solid region with a regular crystal structure. The molten region is extending with increasing temperature, leading to a complete melting of the two selected LiF-KF mixtures at about 1,000°K. In the simulated case, AMoDy reproduces experimental observations with a high degree of accuracy.

FIG. 90. AMoDy modelling of the LiF-KF system phase equilibrium behaviour.

AMoDy was recently extended with a new Force Field model, based on the combination of a pair potential in the form of a Morse potential with a multibody interactions potential based on the Modified Embedded Atom Model (MEAM). It allowed for significant improvement of the modelling capabilities of this code. In particular, now it can model systems that have a very complex crystal structure in solid state, as for instance, ThF₄. The behaviour of these systems cannot be reproduced with a simple Force Field model based on the pairwise interaction between ions in a system.

With the developed Force Field model, the AMoDy code can be applied not only to the structural properties of complex systems, but also to the dynamic properties of molten salt mixtures used in MSR. Figure 91 shows AMoDy results for thermal conductivities of a LiF-ThF₄ system as a function of composition (left) and temperature (right). The results are compared with available recommended values [373]. These calculations provide detailed information about the thermal conductivity dependence on the composition and temperature of a LiF-ThF₄ system. They also complement the experimental information and extend the data to a wider temperature range and the whole range of ThF₄ composition in the salt.

FIG. 91. Thermal conductivity of LiF-ThF₄ system as a function of composition (top) and temperature (bottom). Values calculated by AMoDy and recommended in Ref. [373] are compared.

The viscosity of a LiF-ThF₄ system calculated by the AMoDy code is presented in the left side of Figure 92 for several ThF₄ molar fractions as a function of temperature, and in the right side of this figure for several temperatures as a function of ThF₄ molar fraction. The left side of this figure also include the available experimental [373] and MD calculated [374] values, and

shows good agreement of the AMoDy results. Compared to sparse experiments, AMoDy allows for performing more simulations at different temperatures and composition covering a wider range of parameters and extending available theoretical data.

FIG. 92. Calculated viscosity of LiF-ThF₄ system as a function of temperature (top) and composition (bottom). The temperature dependency calculated by AMoDy is compared with values from [373], labelled as (ref1), and [374], labelled as (ref2).

In general, the results of AMoDy calculations are in good agreement with other published results but provide much more detailed information on the dependencies of molten salts on temperature and on the concentration of specific constituents of the molten salt. This feature is of great importance and shall serve as a source of information on the physical and dynamic properties of salt mixtures applicable in MSRs.

5.12.6. Transient behaviour simulation

The activities grouped in this topical area are dedicated to analysis of system, behaviour and dynamics in nominal and accident conditions. They are driven by the NES internal MSR project and EU projects like SAMOFAR and SAMOSAFER. Two different approaches for MSR transient analysis were developed. One is based on the OpenFOAM solver GeN-Foam [183, 375, 184], which is high fidelity, and the other on the coupled system code TRACE-PARCS [376, 377, 378]. Not all codes can simulate MSR transients because of MSR peculiarities, such as the drift of delayed neutron precursors (DNPs). Also, the heat exchange between volumetrically heated liquids and solid surfaces is a scientifically challenging issue, which can have a strong relevance to safety. The accomplished tasks in this topical area are:

- Development of the GeN-Foam multiphysics solver [183, 375, 184];
- Inclusion of salt properties in the TRACE code and pre-evaluation of the heat exchangers [376];
- Modification of point kinetics in TRACE for DNPs drift [377];
- Modification of TRACE-PARCS for DNPs drift and its application to MSRE transients [378];
- Assessment of decay heat distribution in the fuel stream [379];
- Two MSc studies were finished in 2016. The GeN-Foam solver [380] and the TRACE-PARCS coupled code [381] were preliminary applied to the MSRE and the MSFR, respectively;
- One PhD study related to GeN-Foam application to MSR was launched in 2017 and will end in 2021. It relates to salt freezing phenomena, but covers also designing of an MSR concept [135].

As an example of TRACE-PARCS application, Figure 93 shows results [381] for the steady state distribution of two selected groups of DNPs in the axial cut of the MSFR benchmark geometry. Figure 94 shows another illustrative result [135] from modified GeN-Foam code on a breed-and-burn MCFR design entitled “Tap” reactor. The figure illustrates the impact of baffles in the core on turbulence, where eddy viscosity, as a selected measure, is reduced by at least an order of magnitude in the core with baffles.

FIG. 93. Distribution of groups 2 (left) and 7 (right) of DNPs in the axial cut of MSFR core benchmark geometry [381].

FIG. 94. Impact of baffles on Eddy viscosity in MCFR core. Diagram of baffles (left), Eddy viscosity without baffles (center), and Eddy viscosity with baffles (right) [135].

5.13. R&D ACTIVITIES IN THE UNITED STATES OF AMERICA

The industry has the responsibility to design, construct, and operate commercial NPPs in the United States [382]. However, the US Department of Energy (DOE) has statutory authority under the Atomic Energy Act to promote and support nuclear energy technologies for commercial applications. In general, appropriate government roles include researching technologies with high potential beyond the investment horizon of industry, and reducing the technical risks of new technologies. It is ultimately industry's decision which commercial technologies will be deployed. The DOE role falls more squarely in the realm of R&D. The decision to deploy nuclear energy systems is made by industry and the private sector in economies based on the market.

The DOE Office of Nuclear Energy (NE) mission is to advance nuclear power to meet the nation's energy, environmental, and national security needs. Most US government support for advanced reactors is provided through technology independent, competitive award processes. The overall focus of activities supporting MSR technology by the US government is to facilitate industry success in the deployment of commercial MSRs. Activities by national laboratories focus on developing understanding and technology needed to design, operate, maintain, and evaluate MSRs, as well as providing technical support for safety evaluation tools and methods for commercial development in the near term.

Current activities supporting MSR technology sponsored by the US government include:

- (a) Developing an efficient and effective regulatory process that is technology independent;
- (b) Providing direct support to advanced reactor developers through competitive award processes;
- (c) Providing developer support for access to national laboratories;
- (d) Providing small business innovative research awards to develop MSR technologies;
- (e) Providing support for university research and educational activities;
- (f) Providing innovative, advanced reactor technology development support through advanced research projects agency – energy (ARPA-E) grants;
- (g) Technology development at national laboratories in support of capability demonstration in the fields of safety and safeguards;
- (h) Measurement of (and providing open access to) fundamental fuel salt thermophysical and thermochemical property data;
- (i) International engagement through the GIF and IAEA initiatives;

- (j) Support for development of industry consensus standards;
- (k) Advanced modelling and simulation tool development.

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6. CURRENT CHALLENGES TO DEPLOYING MSRS

This section focuses on the challenges associated with deploying MSRs and possible solutions to these challenges. It is noteworthy, however, that methods or processes for overcoming any specific challenge are mainly incumbent upon decision making by the technology development organization facing the challenge, as well as governments and regulatory bodies. Other stakeholders in the project of an MSR may also support developing these methods or processes and this decision making. Furthermore, many of the issues are specific to the region where the MSR is to be deployed.

The types of challenges that are considered to be the most significant are the following: (i) supply chain, (ii) fuel supply, (iii) regulation, (iv) fuel salt waste disposal, (v) safeguards and security, (vi) maintenance and operations, (vii) component development, and (viii) program documentation. The next subsections discuss each type of challenge. Purposefully, this section does not include the detailed technical challenges associated with MSRs because they are described and discussed within other sections of this TRS. In addition, this section does not discuss existing policy challenges related to deployment of MSRs due to the significant difference in policy positions between Member States. It remains incumbent upon each Member State and organization developing an MSR to assess the policy positions relative to its region of interest to understand the challenges associated with such development in that region.

A possible or potential solution to a type of challenge is discussed at the end of each subsection. Also, recommendations for the IAEA to contribute in that area in the future are mentioned in the final sub-section.

6.1. SUPPLY CHAIN CHALLENGES

Many of the components and structures required by various MSR designs do not have an existing supplier base or full technical specifications necessary for their specialized materials and components. Depending on the regulatory framework, MSRs may be able to rely upon commercial grade high temperature technology for industrial processes. However, some heat transfer components capable of supporting the operating environments presented by an MSR, such as heat exchangers, vessels, and instruments, are currently available in the market. The lack of other components and scarcity of manufacturers lead to the challenges of sourcing new suppliers. This sourcing challenge will also include difficulties in developing, controlling, and maintaining a nuclear safety culture, obtaining technical expertise, and ensuring knowledge transfer across the global supply chain. Additionally, qualification of materials and manufacturing methods is critical, whether nuclear or non-nuclear standards for high temperature are used.

The solution to the supply chain challenge depends on the need or market for MSRs. Since basic manufacturing technology was established for the MSRE in the 1960s (e.g. Ref. [383]), supply chains may be established with a modest effort for deploying current MSRs. That is, if there is a demand, a supplier will appear. This rule does not necessarily work the other way around.

6.1.1. Qualification of suppliers

Inspecting and certifying manufacturing facilities for MSRs is currently difficult if the manufacturer desires to follow an established, conventional nuclear standard such as, for example, ASME Boiler & Pressure Vessel Code (BPVC) [384], which is not directly related to molten salt media. In addition, attempts to use limited capabilities and limited manufacturing facilities may cause long lead times. Risk is increased if required components must come from a link, which has limitations in its capability, within the supply chain. This difficulty associated with qualifying facilities and suppliers is common amongst advanced high temperature reactor developers, especially those that have not chosen to use materials qualified by existing standards.

Because many developers are new entrants into the industry, they have limited experience or capability to qualify a supplier. Developers must be diligent and oversee the suppliers to ensure correct design standards and quality assurance standards are followed. Without fully vetting and qualifying suppliers, the risk of substandard components being supplied is heightened.

The solution for qualification of suppliers also depends on the need or market for MSRs. Taking lessons learned from the MSRE and with participation of current nuclear plant equipment suppliers in the deployment stage, this issue can be resolved. For the suppliers to sell their products, they need to be able to be qualified for this purpose.

6.1.2. Infrastructure considerations

Although infrastructure is generally considered to be regionally specific, the status of existing infrastructure to support MSR deployment presents a challenge. Roads, ports, railways, shipping lanes, heavy haul routes, heavy transportation, and heavy lifting, most of which are project and technology specific, must be adequate. The unavailability of satisfactory infrastructure for a specific MSR deployment project can result in significant challenges to deployment and for meeting schedules associated with critical paths.

The solution for establishing infrastructure could be to utilize the current nuclear plant deployment systems. Besides, deployment of MSRs as SMRs may be facilitated due to smaller and lighter structures and components of SMRs which do not need as sophisticated infrastructure as large nuclear projects.

6.1.3. Reactor developer and supplier engagement

Early engagement between reactor developers and suppliers has long been viewed as critical to success; however, carrying it out requires a delicate balancing act between timing, need, availability, cost, price, competition, and risk. Accomplishing this goal requires obtaining a strategic and cooperative fit between the supply chain strategy of developers and the competitive strategy of the suppliers; an arrangement that is very difficult to achieve. Developers have historically chosen to strategically delay selection of suppliers, whereas suppliers tend to drive towards securing early relationships with the developers. Although the timing, engagement, and decisions made by any particular developer of an MSR will be different for almost every component or service required to deploy its particular design, the need for both the developer and the supplier to be ready for engagement is paramount to success.

The solution to the lack of engagement between developers and suppliers depends on the progress of the deployment of MSR in the market. Since the developers and suppliers are likely based in different countries and probably continents, an international cooperation is required.

6.2. FUEL SUPPLY CHALLENGES

Current developers of MSRs are contemplating a multitude of fuel types and associated suppliers. Fuels include enriched uranium, plutonium from used nuclear fuels, transuranic elements from used fuels, and ^{233}U produced from fertile thorium. Each MSR using liquid fuel will need to have a fuel salt specification to support purchasing. Fuel salt composition and tolerance standards are still being developed.

Most MSRs with fast spectra will require uranium enrichment greater than 5% for their initial cores. Hence, for Member States that currently impose a regulatory limit on the production of LEU to less than 5%, there is a need for regulators and government agencies to consider how to authorize production, transportation, and use of this material. More generally, a strategic plan for MSR fuel cycles showing where and when materials will be generated, transferred, used, stored, and disposed must be developed – just like for any other nuclear fuel.

Since the current fuel supply for LEU and MOX fuels is based on the solid fuel industry, producing molten salt with U/Pu/Th needs new facilities or equipment. Nevertheless, these facilities or equipment are similar to those used in the existing chemical industry. Some MSRs are often compared to chemical reprocessing plants more than to conventional nuclear power plants. The same is assumed as before, that is, if demand exists, supply will appear.

6.2.1. HALEU fuel supply challenges

Available data [385] provides evidence that many MSRs under development are planning to utilize fuel enrichment levels that require HALEU, that is, uranium enrichment $> 5\%$, which is a supply challenge, as described above. An MSR reactor under development planning to utilize less than this enrichment or that is located in Member States where there is an availability of HALEU will not encounter this challenge.

The use of HALEU is potentially problematic in many Member States due to the lack of regulatory requirements for the fabrication, transportation, and use of this type of fuel. Introduction of HALEU for advanced reactors, including MSRs, is being promoted by a nuclear organization in the US [386].

6.2.2. Uranium-plutonium and transuranic (used LWR fuel) fuel supply challenges

A significant investment may be required for developing the necessary infrastructure capable of reprocessing used nuclear fuel from currently operating reactors to make available the required feedstock material for fabrication of the fuel for an MSR. On the other hand, reprocessing used nuclear fuel is expected to reduce the amount of radioactive waste for final disposal. This reduction has benefits in several areas, including financial and environmental benefits.

The solution for utilizing Pu or MA fuels can be limited to the following two cases:

1. For those Member States having commercial reprocessing plants, it may be straightforward by adding only a conversion facility to produce the fuel salt with U or Pu.
2. For other Member States, construction of a new entire reprocessing plant would probably be needed.

6.2.3. Thorium fuel supply challenges

Thorium fuel supply is characterised by different types of challenges than HALEU or Pu. The primary ones remain the acceptance by industry, widespread adaptation of existing uranium-based fuel fabrication technology for manufacturing thorium-based fuel, and the accompanying development of the significant supporting infrastructure for the thorium-based technology. Furthermore, fuels that are based on uranium already have an infrastructure that is well established, so these fuels are a natural preference to less popular thorium. For these reasons, it appears that there are significant challenges for deploying reactor technologies that are based on thorium in the foreseeable future. Nonetheless, if reactor developers increasingly pursue a design utilising thorium, the thorium fuel market will grow.

6.3. REGULATORY CHALLENGES

This TRS recognizes the existence of multiple regulatory agencies worldwide and makes no attempt at evaluating their effectiveness in regulating any reactor technology. Also, it does not identify the challenges faced by any of these agencies in carrying out their individual responsibility to their respective governments, relative to licensing and permitting of nuclear activities involving MSRs. However, in meeting regulatory demands, there are unique, technology-specific challenges associated with licensing of MSR designs irrespective of the regulatory jurisdiction. It is these, more generally applicable challenges from a regulatory perspective that are included here. It remains incumbent upon each individual applicant for regulatory approval to evaluate the specific challenges for licensing or permitting of an MSR within its chosen regulatory jurisdiction. Providing a comprehensive listing of regulatory challenges associated with each of the many regulatory agencies is beyond the scope of this TRS.

The solution to regulatory challenges for the MSR technology may be different depending on each domestic market, but it is desirable to have criteria or guidelines that are applicable worldwide.

6.3.1. Acceptance of evaluation tools by regulators

The scarcity or lack of available technical information to allow developers to create successful designs is a challenge. Hence, providing meaningful and credible evidence to enable the regulator to align or develop its regulatory requirements according to the safety characteristics of MSRs is a challenge as well.

Many of the issues impeding industry interest in MSRs are in the areas of safety evaluation and regulatory licensing tailored to the characteristics of MSRs. Consequently, common safety/licensing challenges exist from the lack of specific safety evaluation tools and capabilities. The outstanding issues are, among others, lack of a consensus set of potential MSR

accidents, of the form and content of a regulatory license application, and of experimental validation or benchmarks of safety evaluation tools for MSR. Moreover, the current regulatory practices caused the acceptance of the newly developed tools to be very expensive and time consuming.

6.3.2. Evaluation of accident progression

Depending on the regulatory framework, licensing of an MSR may require evaluating the progression of potential accidents for MSRs. No comprehensive, consensus set of potential MSR accidents or safety models of these accidents are currently available. This lack of documented and credible MSR accident scenarios that result in the dispersal of significant quantities of radionuclides may be due, at least partly, to the good safety characteristics of MSRs. A robust justification of these safety characteristics is expected to completely or at least partially resolve this challenge.

However, it is unlikely that tools for modelling accidents for MSRs will need to be highly precise as this kind of NPP may be able to trade its very large safety margins against modelling precision. Overall, a comprehensive set of DBAs could be developed for MSRs based upon their common characteristics in a similar way to that initially developed for LWRs, well before the latter plants had accumulated very considerable operating experience.

MSR technology offers great flexibility to reactor developers, so there are substantial design variations between prospective developers, thus increasing the difficulty of validating modelling tools for general purposes. Moreover, MSRs will have substantial variability in types and vulnerabilities to accidents that occur beyond the reactor system and the generation of electricity and possibly cogeneration. However, these accidents may also lack the energetic potential to disperse the radionuclides beyond the plant site, owing to the inherent properties of molten salts and low operating pressure of the primary system.

One means to provide assurance that a proposed list of DBAs for MSRs adequately spans the potential set of accidents is to employ maximum hypothetical accident evaluations such as those used by the US regulator [387]. This approach led to historic (and current non-power) reactor licensing that relied upon maximum hypothetical accidents to bound potential accidents. MSRs may be able to follow a similar strategy by showing that even credible beyond DBAs (BDBAs) such as prompt criticality or massive failure of the boundary wetted by the fuel salt or both do not result in significant radionuclide release while crediting only barriers and passive structures, systems, and components (SSCs) for removing decay heat.

Measurements of fuel salt's properties are required to provide data to determine, in subsequent testing, what radionuclides are released from the fuel salt under normal and accident conditions. The combination of fuel salt property measurements and performance models will form the foundation for fuel salt qualification.

Fuel accidents can also be bounded by maximum hypothetical accidents. If even prompt criticality or unintended criticality outside the core region of the fuel boundary does not result in release of radionuclides from the plant, the fuel performance specification and consequently the testing required to validate fuel properties may be relaxed substantially. In essence, even if fuel salt that is well outside design specifications is loaded into the reactor, the probability for radionuclide release into the environment remains very low provided that the fuel continues to be cooled, and some MSR designs provide this cooling by passive means. Thus, a fuel specification would be limited to validating the fuel salt's thermophysical properties rather than

its nuclear properties. MSR fuel salt thermophysical properties are not anticipated to substantially change even after many years of use, thereby substantially decreasing the rigor required in the measurements of these properties [388]. These are benefits of using a fuel salt that may simplify the licensing process.

Additionally, any accidents must be evaluated that might happen during the entire fuel cycle and also after the used nuclear fuel has been disposed of. A regulator expects any Member State to support their safety case. An ultimate waste disposal plan shall be prepared, and its safety reasonably demonstrated by the license applicant.

6.3.3. Industry consensus standards

MSR deployment challenges exist due to a lack of industry consensus standards, several of which have begun development. Examples of recommended standards include:

- MSR Design Safety Standard. For example, one is currently under development in the US as ANS 20.2;¹⁷
- salt composition standards;
- fuel salt transportation standards;
- standard methods for performing MSR fissile materials accounting; and
- standard methods for evaluating MSR vital areas related to physical protection.

The solution to prepare licensing standards is now starting in the US, such as ANS 20.2. However, there is a wide variety of these standards besides the above examples, and further development is needed.

6.4. FUEL SALT WASTE DISPOSAL CHALLENGES

Liquid salt does not accumulate physical damage like solid fuel. Also, to the extent that its composition can be controlled to remain within a specification, it does not suffer from aging. Fission products can potentially build up to equilibrium concentrations and most designs include some online or batch continual chemical processing (polishing) of the molten salt so the extent of this phenomenon will be dependent upon the individual developers' choice of fission product and fission gas management techniques and design choices. Consequently, used MSR fuel salt can be considered for use in future generations of reactors. However, further advances in technology development and demonstration are required for an MSR to operate on a closed fuel cycle [116]. Nevertheless, MSRs still require a disposal path for fuel salt waste and defining this path could significantly support licensing of the MSRs.

The Office of Nuclear Energy of the US DOE has recently provided overviews of the potential waste processing and waste form options for MSRs [389] and has evaluated the status of molten salt waste technologies focusing on fast spectrum systems [390]. It is still required, however, that thermal spectrum reactors be evaluated, and that challenges in development of disposal techniques of different fluoride and chloride salts be solved.

¹⁷ American Nuclear Society, Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten-Salt Reactor Nuclear Power Plants, ANS-20.2, <https://ans.org/standards/involved/voloppor/rarcc/advanced/>

The basic technology for creating stable salt waste forms is generally known [391, 392]. However, no implementation details are available for the technology or licensing of waste disposal systems. A validated MSR waste stream roadmap, which would help to minimize the technical risks, is yet to be completed.

Some components wetted by fuel salt may have significant amounts of fission products plated onto and possibly embedded into their surfaces. Moreover, container alloys will have been activated while in use and may include surface contamination with actinide materials. Activation of nickel within the nickel-based alloys used may (depending on the shielding design) result in the structural alloy being classified as a radioactive waste and disposed of. Several Member States are working towards developing nuclear waste disposal strategies that, in principle, would be the same for MSR and LWR waste. Also, recycling of used nuclear material is possible.

The knowledge of the waste streams and viability of solutions to treat the used fuel and dispose of the waste are important aspects in developing and licensing MSRs. There are technical options for reusing actinides indefinitely and for stably packaging the remainder of the fuel salt. The remainder of the solid wastes can largely be left to decay sufficiently to be acceptable in surface storage sites.

6.5. SAFEGUARDS AND SECURITY CHALLENGES

MSRs incorporate features that are fundamentally different from conventional reactors. At the same time, MSRs have limited operational experience restricted to research reactors from a time preceding modern best practices for safeguards and security. A modern framework for safeguards and security for MSRs is therefore largely untested and represents important uncertainties regarding deployment of MSRs as power reactors at large scale. Consequently, reactor developers must establish, verify, and validate adequate safeguards and security provisions for MSRs. In parallel, regulatory stakeholders - including legislators, international organisations, and national regulators - will need foresight, imagination, and initiative to proactively accommodate an expansion of nuclear power at large scale, including SMRs and MSRs, as prescribed by many scenarios for reducing greenhouse gas emissions [393].

There is great variety in MSR designs which importantly impacts safeguards and, to a lesser extent, security. Some MSRs circulate liquid fuel salts while actively controlling the chemistry of the fuel salt, which are features resembling facilities for reprocessing spent nuclear fuel. This is particularly true for MSRs that incorporate online refuelling and fuel salt let-down, as well as online separation of fission products and actinides. Other MSR designs do not exchange considerable amounts of radioactive material with other systems within the NPP, more akin to conventional solid fuel reactors, for which nuclear material accountancy is expected to be easier to implement.

Historically, safeguards and security measures were 'added on' rather than developed concurrently with the first NPPs. Despite this, the safeguards and security measures implemented in conventional NPPs have been effective. This is evident from the lack of severe incidents related to external attacks, sabotage, theft, or other diversions of fissile material from commercial NPPs. In designing and developing commercial MSRs, proactive consideration of and discourse about safeguards and security from an early design stage will ensure effective implementation of efficient measures that also retain overall economic competitiveness.

Even though the current legal framework for IAEA safeguards [394] focuses mostly on existing NPPs and processing facilities, already established methods and practices are expected to be applicable to MSR to some extent, though some adaptation may be necessary. Nonetheless, since no MSR for power generation has been licensed for commercial operation in any regulatory jurisdiction to date, the implementation of necessary safeguards and security protocols remains a challenge to deployment.

6.5.1. Material control and accountability

Fissile material control and accountability shall be a key differentiating element of licensing risk for reactors with liquid fuel. Some Member States plan to adopt Material accountancy equivalent to what is required currently by The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons [395]. A new approach, such as including safeguards early in the design process, can be implemented as well.

Depending on a reactor's design, the application of safeguards to MSRs with liquid fuel may have to consider: (i) the mixture of fuel, coolant, fission products, and actinides (along with high radiation fields); (ii) variation in isotopic concentrations in the fuel salt including removal of fission products, rare earth elements, and noble metals; (iii) high operating temperatures; (iv) online processing – if included in the reactor design – where some fraction of the inventory of fuel salt can be removed while the reactor is operating; (v) unique refuelling schemes – if included in the reactor design – including the ability to continuously feed the core with fresh fissile and fertile material; and (vi) the presence of fuel outside the reactor vessel [396]. Approaches to accomplish the required material control and accountancy at MSRs are still under development and no general consensus has been reached to date.

Nevertheless, there are already existing safeguards techniques and equipment, classified by the IAEA [397], that could be directly implemented at MSR facilities, such as the weight or volume counting of the fuel salt within a single Material Balance Area (MBA), among other non-destructive analysis techniques. With only one or limited number of MBAs encompassing the MSR NPP, material control and accountancy can be achieved in a simplified manner, which would correspond to trends in advanced reactor design where a simplified approach to safety and safeguards is preferred to costly and redundant engineered solutions. However, the fact that even fuels for a pebble bed reactor (PBR) could be seen as a bulk item versus an item when looking at reactor design and operation complicates the accounting for fuel moving across MBAs [398]. Measuring and accounting for fuel in pebble and liquid or even the solid salt forms will need a change in the framework for verification and accountancy measures [399]. The safeguards will have to acknowledge that liquid material changing isotopic form cannot be drawn off from an MSR and be accounted like an item. A format for setting up a limit on the material measurement accuracies and discrepancies such as is done with powders and liquids in fuel fabrication or reprocessing facilities will need formulation for MSRs and PBRs.

Additionally, containment and surveillance techniques, which complement measurements, do not necessarily have to be different from the existing ones. If safeguards are considered early in the process of designing an MSR, complexity of accountability processes can be significantly reduced. Solid fuel salt delivered to an MSR NPP prior to melting can be treated as fuel pellets currently used in LWRs and counted per item. Transport and storage accountancy will not differ, either. On the other hand, transport and handling of the nuclear material will

increase due to rapid development and deployment of MSR and closed nuclear fuel cycles, which will require infrastructure and more personnel capable of dealing with sensitive technology and with the noted need above for a more sophisticated approach to handling bulk nuclear fuels with acceptable uncertainties in verification activities.

6.5.2. Security challenges unique to MSRs

Security by design, a concept included in the World Institute for Nuclear Security (WINS) Best Practice Guide [400], suggests that prevention of a malicious act is always preferred to having to respond to it. Design and salts that are proliferation resistant, lessons learned from operation of existing NPPs, and strict access control of the MSR NPP shall be implemented throughout the entire lifetime of an MSR NPP.

The only anticipated MSR interface with the environment that is related to security is in the MSR's need to reject decay heat – which is the same as in conventional NPPs. The function of removing decay heat after reactor shutdown will likely be split into multiple passive (and active) systems to provide redundancy against single failures. To cause sufficient barrier damage to result in significant release of radionuclides into the environment by preventing an MSR from rejecting decay heat, an adversary would, thus, need to impair these separate and independent cooling systems and prevent operation of the cooling systems used during normal operation. Furthermore, molten salts retain radionuclides to a much greater extent than when traditional solid fuel rods are used. Hence, MSRs do not appear to be particularly vulnerable to releasing radioactivity due to attacks to systems used for decay heat removal.

On the other hand, MSRs have various SSCs that are different to those used by the current fleet of NPPs. The SSCs that are specific to MSRs, such as those used for online fuel reprocessing, may be targets of malicious attacks. Accordingly, measures for the physical protection of these SSCs may have to be developed, and then reviewed for regulatory approval. Such development and review are challenges to be addressed and are related to the regulatory challenges mentioned above.

Alternate physical security requirements for advanced reactor technologies with enhanced engineered safety and security features, such as those the Nuclear Energy Institute (NEI) in the US has proposed developing, are needed [401]. The alternate requirements would have the key result of shifting the responsibility for protecting facilities from the plant owners to local law enforcement, substantially reducing the ongoing expense related to security by the plant owners. SSCs that are specific to MSRs, such as those used for online fuel reprocessing and the interface for transferring decay heat to the environment, will need to be robust against assaults so local law enforcement can be relied upon for their physical protection.

6.6. MAINTENANCE AND OPERATIONS CHALLENGES

The MSRE, which was operated and produced energy just below 10 MW(th) for roughly four years, is a major source of knowledge on maintenance and operation of an MSR. There are elaborate reports on maintenance equipment and procedures, and accounts of successful implementation of totally remote and long-handled tools for maintenance of the MSRE, including a mock-up facility of an MSR specifically designed for testing of remote handling technology for major component replacement [109], [402]. Remote maintenance is required due to the high temperature and radioactivity of the fuel. Harsh conditions inside an MSR facility can damage the remote-handling equipment, which adds another complexity level to the maintenance and operations, as service of the irradiated equipment might be needed as well.

Moreover, the operational challenges in decontamination and cleanup of spillages of active salts shall be addressed by the Member States. Also, should any in-core structures and components be replaced, either due to a break or their end-of-life caused by irradiation, specific maintenance actions are required. Measures used in conventional NPP can be adopted in MSRs to some extent, but a more advanced and remote solutions might be needed, depending on the strategy for dealing with the decontamination and replacement.

Improved modelling tools for process operations would be of substantial use to facilitate scale-up and modelling of MSR operation for extended duration. Extensive, automated, and remote maintenance and component replacement may be required for a commercial plant. Developing and demonstrating this maintenance and replacement, as well as adequate modelling and simulation of maintenance and operation processes are key elements and challenges to deploying MSR designs. Both the maintenance and operation of an MSR could be significantly simplified with the use of modern technology, such as virtual reality, where a digital twin of a facility could be used for training and planning of the activities. However, challenges that may be faced during maintenance and replacement of the remote handling equipment will require solutions that might not yet be commercially viable.

6.7. PROGRAM DOCUMENTATION CHALLENGES

The MSRE was viewed as a very successful test reactor and the overall MSR program of the US was a development program that extended over decades, which would suggest that no major technology development is needed in order to commercialise the first MSR NPPs. However, this understanding is not widespread and the technology status and the unique features and benefits of MSRs are not well known beyond the people who have been involved in various MSR development programs.

For example, corrosion of pipes and other components in contact with salts has been extensively researched and documented. Few people outside of a very narrow specialist community, including regulatory agencies, understand that, as early as in 1967, the ORNL concluded the technology for materials resistant to corrosion was in an advanced stage of development [403]. ORNL came to this conclusion based on an extensive corrosion program, utilizing several families of fluoride mixtures involving commercial and high temperature alloys that were under development at that time. These same studies also concluded that container materials were available that demonstrated extremely low corrosion rates at temperatures expected to be encountered during anticipated MSR operations [404].

Through the 1970s, salt chemistry control was further developed, substantially decreasing the significance of corrosion as a limitation to MSR deployment. Also, multiple companies are currently carrying out their own corrosion experiments.

Irrespective of these examples of research and development results, which have been available for several decades, there remains a lack of general familiarity with the technology status of MSR developments and the unique features and benefits of MSRs. Developing program documentation of high quality that is readily accessible to the non-specialist technical community is more important to MSR development than for other advanced reactor classes.

6.8. RECOMMENDATIONS

Near-term deployment of MSRs is potentially viable as extensive R&D activities have been undertaken in several Member States. The knowledge gained from the MSRE has been used and progress has been achieved with modern research and technology. As a result, various advanced designs have been proposed and development thereof continues to be pursued. However, several challenges to commercial deployment remain unresolved.

Addressing the challenges mentioned in this section could be facilitated by the joint effort of the international community involved in MSR technology development. In particular, the following challenges have been identified that could be better addressed at an international level and especially with the IAEA providing a discussion opportunity to Member States developing MSRs: (a) reactor developer and supplier engagement; (b) HALEU fuel supply; (c) uranium-plutonium and transuranic fuel supply; (d) safety challenges, including preparation of guidelines; (e) fuel salt waste disposal; (f) material control and accountability; (g) security; (h) maintenance and operation; and (i) program documentation.

7. SUMMARY AND CONCLUSIONS

MSRs are a broad and diverse category of reactors, in which a molten salt plays one or more significant functions in the reactor core, such as fuel chemical form, liquid fuel carrier, coolant, or moderator. This section provides a summary and offers the main conclusions regarding the status of the technology of this kind of reactor.

7.1. HISTORY OF MSR TECHNOLOGY

The basic elements of MSR technology were developed by the US government through the aircraft nuclear propulsion program and later the molten salt breeder reactor program from the 1940s to the 1970s. Several other Member States developed complimentary programs after publication of the work at the second UN Atoms for Peace conference in 1958. Two technology demonstration reactors were built and operated in the 1950s and 60s along with numerous in-pile test loops and critical facilities. ORNL's MSRE had remarkably high availability and smooth operational performance, demonstrating the overall soundness of MSRs as a reactor concept. The engineering challenges associated with MSR technical viability were largely overcome by the mid-1970s. Conceptual designs that do not involve creation of separated streams of fissile materials were proposed in the late 1970s. MSRs, however, have not received the level of financial backing necessary to proceed into large-scale development and deployment. The chasm between technical viability and viable deployment has yet to be bridged. Modern efforts to revive the technologies build substantially from the extensive documentation developed under the historic program.

7.2. ADVANTAGES AND TECHNICAL CHALLENGES OF MSR TECHNOLOGY

Section 3 presents an overview of MSR advantages and challenges for deploying MSRs. The Member States' perspective on the topic is found to be in good agreement, despite the numerous designs that qualify to be an MSR. Importantly, no fundamental scientific barriers are identified to impede the development of MSRs, while a few major technological challenges are still to be solved prior to the technology scale-up and deployment.

The successful operation of the MSRE in the 1960s demonstrated the feasibility of the technology and provided a massive source of information that current developers are drawing from, crucial to fast-track the technology development. Moreover, developers benefit from the substantial scientific advancements achieved in almost every technological field since the MSRE, which contribute to establishing MSRs within the modern energy supply and regulatory frameworks and reduce the risk of failure in commercial deployment.

The numerous intrinsic advantages of liquid fuels and near-ambient pressure coolants cannot be overestimated. The novel technological challenges that are encountered in materials qualification, which arise due to the extreme radiation environment and the far higher temperatures than those in WCRs, only partially shadow the promise of a reactor design that could revolutionize the nuclear energy landscape. The compact form factor and modularity of several designs open to possibility of decentralized nuclear power production, a tectonic shift in the sector.

To reach suitable power density and reactor lifetimes to make MSR's a market-driven alternative to fossil fuel power generation, the following are regarded as the major technical challenges ahead: i) the development of modelling software and databases of salt thermophysical properties that are acceptable by a regulatory body; ii) the mitigation of fast degradation of the core's structural materials induced by the harsh corrosivity of molten salts and the extreme radiation environment, and iii) design and manufacturing of components (e.g. heat exchangers, pumps, and valves) capable of operating in such extreme environments.

Many — if not all — of the technical challenges ahead of MSR's will benefit from tighter collaboration between public and private entities. The knowledge and experience of universities and national laboratories could be coupled with the ability of commercial developers to attract venture capital, fostering a virtuous circle of collaboration and mutual benefit which will hasten the deployment of this technology.

7.3. CLASSIFICATION OF MSR FAMILIES

There are many parameters and options on how to develop an MSR taxonomy. Similar to an animal taxonomy, some of the parameters may or may not be important for its structure. All historical and recent concepts rely either on fluoride or chloride salts. Accordingly, other halides or other salts are not considered because one or more of their properties disqualify them for use in the reactor core. Furthermore, the R&D effort behind the MSR concepts, as well as the respective TRL level, strongly differ.

In the past, there were several efforts to classify MSR's, for instance by the location of the first heat exchange between fuel and coolant. The taxonomy developed in this document follows the major reactor types and technological similarities and is described in Section 4. In a simplified way, the MSR's are divided into three major classes according to the type of materials present in the core. Accordingly, the active core of the first class includes graphite and salt, the second class rely solely on the salt, and the third class includes structural material, which is separating the fuel from the coolant or moderator. This division into three major classes assure that there is certain technological similarity within each class (see Figure 95).

The first class, Graphite based MSR's (I), is characterized by graphite presence in the core, and employ exclusively fluoride salt and thermal spectrum. Since the graphite is directly compatible with the salt, there is no need for other structural material in the core. This class has the highest TRL level, because of the successful operation of the MSRE, and for some concepts because of their similarity to High Temperature Reactors (HTR's).

The second class, Homogeneous MSR's (II), is characterized by the absence of structural materials, whose function is to separate the fuel salt from a coolant or a moderator in the core, which is thus filled solely by the fuel salt. Optionally, there can be structural materials (e.g. baffles for flow direction) in the core; however, they do not have the separation function. The TRL of this class depends mainly on the fuel salt, and for some of the MSR's using similar salts to those of the MSRE or MSBR, the TRL can be higher than for others.

The third class, Heterogeneous MSR's (III), is characterized by the presence of structural material in the core that is separating the fuel salt from a dedicated coolant or from a moderator not directly compatible with the fuel salt. The TRL of this class depends also on the separation

material and can be lower than for the previous two classes because these materials are typically not yet qualified.

FIG. 95. First three layers of MSR taxonomy (circle view).

One more class, Other MSRs (IV), includes concepts that do not fit the other three major classes, and that generally have the lowest TRL.

Each class is further sub-divided into families and types. Each of the major three classes (I, II, III) has two families, and hence there are in total six major MSR families. The families of the Graphite-based MSRs (I) are identified according to the location of the fuel as Molten salt cooled reactors (I.1) and Graphite moderated MSRs (I.2). The class of Homogeneous MSRs (II) is characterized by a core filled solely by the fuel salt and a fast neutron spectrum. The two major families are determined by the salt type: Homogeneous fluoride fast MSRs (II.3) and Homogeneous chloride fast MSRs (II.4). The third class, Heterogeneous MSRs (III), is characterized by the presence of structural material in the core that is separating the fuel salt from a dedicated coolant or from a moderator not directly compatible with the fuel salt. The two families in this class are defined by the material that needs to be separated. In the Non-graphite moderated MSRs family (III.5), the moderator is not directly compatible with the fuel salt, and in the Heterogeneous chloride fast MSRs family (III.6), it is the dedicated coolant that is not directly compatible.

Within each family, the types are defined by rather different criteria, which are specific to each family, for instance, by solid fuel shape, fuel cycle type, moderator state, or coolant type.

In the case of the Other MSRs class (IV), families are not considered because the MSR concepts are different from each other, and types are presented directly. These types are typically historical or discontinued. If preferred, they can be reclassified into the three major classes based on the materials present in the core. There are also concepts that are on the edge between two classes. For example, the graphite moderated MSRs with two salts separated by structural material are included in class (I).

7.4. RESEARCH AND DEVELOPMENT ACTIVITIES

The flexible and risk informed regulatory framework adopted by the CNSC is suitable for licensing advanced reactors. Two small modular MSRs, IMSR by Terrestrial and SSR by Moltex are among a range of SMR designs under the pre-licensing VDR, which is an optional service in three phases to assess new reactor designs against CNSC design requirements. In parallel, experienced Canadian operators of generation of nuclear power are working with SMR developers to advance engineering and design work and evaluate potential demonstration projects. IMSR by Terrestrial is one of the SMR designs selected by Ontario Power Generation (OPG) and SSR by Moltex is one of the SMR designs affiliated with New Brunswick (NB) Power.

Several MSR research projects were launched in 2016 at CNL's CRL site. The research programme is intended to develop modelling capabilities and experimental facilities to generate

data required for deployment of small modular MSR. The research projects address areas related to thermophysical properties of coolant and fuel salts, corrosion of materials in molten salts, multiphysics modelling for safety and performance, and evaluation of waste streams and safeguards challenges of MSR. In addition to AECL and CNL, MSR R&D activities are also performed in several Canadian universities, including the University of New Brunswick.

In January 2011, SINAP, CAS restarted a research program on thorium utilization named TMSR (Thorium Molten Salt Reactor) program. This program focuses on the research and development of technologies supporting the TMSR to achieve safe and economic use of Generation IV MSR NPPs and highly efficient use of the thorium resource. In the last ten years, it developed key TMSR technologies, such as molten salt production and purification, an alloy that is corrosion-resistant (GH3535), superfine grain graphite, molten salt technology for corrosion control, and molten salt loops and their components (pump, heat exchanger, flowmeter, valve, reactor vessel). These technologies were used in the construction of a mock-up reactor, named TMSR-0, which is used for activities such as testing, verifying, and licensing technologies, and training of reactor operators. Construction of TMSR-0 was finished in 2019.

A 2 MW(th) experimental reactor, named TMSR-LF1, is being developed. Site selection, environmental impact assessment, basic design, preliminary engineering design and construction drawing design of this reactor were completed. In December 2018, the NNSA approved the Hong Sha Gang Industrial Park in Wu Wei, Gansu Province as the site for this reactor. On March 2020, the construction of the TMSR-LF1 plant officially began, and the plant's main building was completed by the end of 2020. The equipment for TMSR-LF1 has been manufactured, and installation started by the end of 2020. For future application, a 100 MW(e) SMR named smTMSR-400 is being designed as a thorium convertor and in situ burner driven by low enriched uranium. It can be used for electricity generation, and to satisfy diversified energy demands.

MSR technology has been developed in the Czech Republic since 1999. The national MSR Technology Development Program focuses primarily on experimental research and verification of selected parts of the technology in the areas of MSR physics and fluoride salt neutronics; development of the thorium-uranium fuel cycle technology, including the pyrochemical methods for the separation of actinides and fission products from a molten salt; research on materials; and development of selected components for an MSR. This program is supported by the Ministry of Industry and Trade and the Technology Agency of the Czech Republic. The main research organization responsible for the domestic development of MSR technology and for international cooperation is the Research Centre Řež. The main goal of the program is to significantly contribute to the worldwide development of this technology.

In Denmark, work on MSR technology is conducted at a university (DTU), Copenhagen Atomics, and Seaborg Technologies. While DTU is focused on modelling and simulation, the two companies are developing their own MSR equipment and reactor designs that are planned to be commercialised in mid-2020s. The Danish contribution to the MSR community comprises R&D activities such as coupling schemes of neutronic-thermohydraulic codes, design of molten salt loops and freeze valves, and development of alternative MSR designs.

The JRC of the European Commission has the following capabilities: Synthesis of actinide fluorides and chlorides; purification techniques and purity characterization methods for molten

salts; investigations of phase diagrams for actinide (and non-actinide) fluoride and chloride systems; electrochemistry of fluoride and chloride salts; Raman spectroscopy of solid halides with deployment in the short term of measurements of liquid salt; development in the short term of density measurement of liquid halide salts; measurements of thermal conductivity of solidified halide salts with deployment in the short term of measurements of liquid halide salts; main developer and owner of the JRCMSD thermodynamic database on halide systems; Post Irradiation Examination of nuclear fuels, including MSR irradiated fuels; development and design of structural components for irradiation purposes; and Raman spectroscopy technique for determining oxide impurities and molecular species stabilized in the liquid phase of fluoride and chloride salts.

The JRC also provides determination of: Melting points of MSR fuel and coolant salts; vapour pressure of MSR fuel and coolant salts; boiling points of MSR fuel and coolant salts; fusion enthalpy and enthalpy of transition of MSR fuel and coolant salts; heat capacity of MSR fuel and coolant salts; fission product release/retention for MSR fuels; solubility limits for noble gases, such as He, Kr and Xe, in molten salt fuels; and mixing enthalpy of multi-component halide systems.

The research in the European Union is carried out in the framework of national research programmes combined with the European Framework R&D programmes. It evolved from research on the historical ORNL concepts to the MSFR proposed by the CNRS. The research aims to understand, model, and design all physics and chemical processes in the fuel circuit of the reactor and in the fuel salt processing stages, and to assess the sustainability and safety of the reactor design. The research is based on experimental work, and on advanced numerical modelling combined with experimental validation. Various experimental setups at large scale were designed to measure 1) the physicochemical and thermophysical properties and the behaviour of the (fuel) salt under a wide range of conditions, and 2) the dynamic properties of internally heated loops. Section 5 describes detailed results provided by European Member States.

In France, CNRS has been involved in molten salt reactors since 1997. An innovative concept called Molten Salt Fast Reactor (MSFR) was proposed, resulting from extensive parametric studies in which various core arrangements, reprocessing performances, and salt compositions were first investigated to adapt the reactor in the framework of the deployment of a reactor fleet based on thorium on a worldwide scale. The primary feature of the MSFR concept is the removal of the graphite moderator from the core, resulting in a breeder reactor with a fast neutron spectrum and operated in the thorium fuel cycle. This “reference MSFR” (large power reactor operated in the Th fuel cycle) has been recognized as an alternative for the long term to fast neutron systems with solid fuel, and with unique potential (such as negative reactivity coefficients, smaller fissile inventory, easy in-service inspection, and simplified fuel cycle). Accordingly, it was selected for further studies by the GIF in 2008. A safety analysis methodology for such a reactor with circulating liquid fuel has been developed in collaboration with IRSN, Framatome and Politecnico di Torino in the frame of the SAMOFAR and then in the SAMOSAFER European project of the H2020 program, including now also CEA and Politecnico di Milano.

The French R&D activities described in this document are supported by the development of innovative simulation codes and by experimental facilities, both dedicated to MSR technology. The research activities cover reactor physics including neutronics and transient calculations, experimental molten salt chemistry, risk analysis and evaluation, decay heat calculation, structural material corrosion, plant design, and experimental technology for molten salt. New studies are being conducted at CNRS/IN2P3 on other kinds of MSFR, for example as an SMR of reduced power, or operated in the U/Pu fuel cycle, or as an actinide burner, or a combination of these characteristics. These research activities are done in collaboration with an increasing number of French partners (Orano, CEA, Framatome, IRSN, EDF), which reflects a growing national interest for MSR technology in France during recent years.

The R&D activities in Italy started in the context of the first European projects on the development of MSR technology (e.g., ALISA and EVOL), and have continued during the last ten years (2010 – ongoing). The institutions that are mainly involved in the effort are two universities, Politecnico di Milano and Politecnico di Torino, mainly focusing on neutronics, thermohydraulic, and control aspects, and in general on modelling and simulation. The Italian institutions developed several modelling tools specifically for studying the peculiar features of a molten salt system. Politecnico di Milano built a testing facility (DYNASTY) aimed at the experimental investigation and model validation of the natural circulation in presence of distributed heating generation.

Based on the ORNL's results on MSR technology obtained up to the 1970s, Japan has been investigating the MSR "FUJI", which uses fluoride fuel salt and is moderated by graphite since the 1980s. The following MSR designs have been developed: FUJI-U3 (achieving a 1.0 conversion ratio by using ^{233}U -Th fuel salt), FUJI-Pu (using Pu as start-up fissile material, and transmutation capability of minor actinides), the super-FUJI (generating 1,000 MW(e)), and a mini-FUJI as a pilot plant. Since 2019, the Japanese government started to support the development of MSR technology, and three MSR venture companies are selected; two are promoting an MSR with fluoride salt that is moderated by graphite, and one is promoting a fast MSR with chloride salt.

The research in the Netherlands is carried out at NRG in Petten and at TU Delft. At NRG fuel salt samples have been irradiated (SALIENT experiments) in the High Flux Reactor and the first ones have been unloaded and prepared for post-irradiation analysis at the laboratories in Petten and at the JRC-Karlsruhe. Besides the ongoing irradiation of fuel salt samples, alloys based on nickel have been prepared for irradiation to study helium embrittlement in these materials. In the calculation field, the SPECTRA thermohydraulic code originally designed for LWRs was extended to be used for MSRs. The code can be used for the assessment of various operational transients and accident scenarios. NRG and TU Delft jointly carry out experimental and calculation studies on flotation and the extraction (by gas bubbling) of insoluble particles like noble metals.

The research at TU Delft focuses on the development of an advanced multi-physics numerical code system for transient analysis based on in-house developed discrete-ordinates and fluid dynamics codes. To reduce computation time, non-intrusive ROM was developed and applied to uncertainty analysis in steady state. Afterwards, the methodology was extended to transient scenarios of the MSFR where time is treated as any other input parameter. Besides the ROM models, GPC expansion has been applied to uncertainty analysis of the MSFR. GPC is of

particular use in case of strong non-linearities. CALPHAD assessments, some of which are coupled to GPC, have been used to develop thermodynamic models and calculate the thermodynamic properties of salt systems, including the fuel matrix, various fission products (especially caesium and iodine), and the corrosion products resulting from the interaction of the salts with structural materials (nickel alloys). More fundamental structural studies of molten salts using EXAFS spectroscopy interpreted with molecular dynamics studies give insight in the structural characteristics of the molten (fuel) salt. The aim is to gain predictive power about the thermodynamic behaviour of the salt.

Over the last twenty years, R&D on MSR in the Russian Federation has been mainly focused on fast spectrum concepts with or without thorium support. These concepts were recognized by GIF as an alternative for the long term to fast neutron reactors with solid fuel, and with attractive features, namely strong negative reactivity feedback coefficients, smaller fissile inventory, and simplified fuel cycle. The MOSART concept can operate within technical limits using an alloy with high nickel content as container material, and with different fuel loadings and make-up based on TRUs (from used LWR fuel with MA/TRU ratio up to 0.45) as special actinide transmuter, as self-sustainable system (conversion ratio (CR) ≤ 1), or even as a breeder (CR > 1). The Russian Federation established an experimental complex in support of future R&D on MOSART. The test reactor facility consisting of a 10 MW(th) MSR (LiF-BeF₂-A_nF_n fuel salt) with a homogeneous core is currently being designed.

The MSR R&D activities in Switzerland concentrate within the domain of the Swiss Federal Institutes of Technology. They are carried out predominantly at PSI in cooperation with the universities ETH Zurich and EPFL Lausanne through students' projects. Historically, there was a dedicated MSR research project already in the 1970s and 1980s at the PSI predecessor, EIR. In the early stage, it focused on the possibility of using aluminium tri-chloride (AlCl₃) salt as a direct boiling coolant for a homogeneous chloride fast reactor and for a solid fuel fast reactor. Later, several options of a fast heterogeneous MSR cooled by salt were considered, including the option to use blanket salt as the primary coolant. The latest studies of the early stage were dedicated to homogeneous chloride fast MSRs with natural chlorine isotopic composition. Nowadays, the research at PSI focuses on MSR safety and sustainability. It covers a wide range of different topics, which rely on simulation and modelling. The major tools developed at PSI, also applicable for MSR simulation, are equilibrium fuel cycle procedures EQL0D and EQL3D, GEMS for thermodynamic modelling, AMoDy code for molecular dynamics, and multi-physics tool GeN-Foam for MSR core analysis. The research is not focused on a particular MSR concept, and it has a rather education and monitoring character. The MSFR is considered as the reference design. However, the fuel cycle performance of fluoride salt cooled reactor, graphite moderated MSR, homogeneous fluoride and chloride fast MSR, and non-graphite moderated MSR was assessed. Due to the lack of design data, the heterogeneous chloride fast MSRs were not included so far. The major focus of the ongoing research is the safety of MSFR, and conceptual fuel cycle and thermohydraulic design for a Homogeneous B&B chloride fast reactor.

Current activities supporting MSR technology sponsored by the US government include: Developing an efficient and effective regulatory process that is technology independent; providing direct support to advanced reactor developers through competitive award processes; providing developer support for access to national laboratories; providing small business

innovative research awards to develop MSR technologies; providing support for university research and educational activities; providing innovative, advanced reactor technology development support through ARPA-E grants; technology development at national laboratories in support of capability demonstration in the fields of safety and safeguards; measurement of (and providing open access to) fundamental fuel salt thermophysical and thermochemical property data; international engagement through the GIF and IAEA initiatives; support for development of industry consensus standards; and advanced modelling and simulation tool development.

7.5. CURRENT CHALLENGES TO DEPLOYING MSRS

Section 6 discusses non-technical challenges to MSR deployment. Although the MSR technology has been under active development, there are still significant obstacles to overcome before MSRs can be commercialised. The identified challenges are:

- supply chain challenges, including qualification of suppliers, infrastructure, and developer–supplier engagement;
- fuel supply challenges, including HALEU fuel supply, used LWR fuel supply, and thorium fuel supply;
- regulatory challenges, including acceptance of evaluation tools by regulators, evaluation of accident progression, and industry consensus standards;
- fuel salt waste disposal challenges;
- safeguards and security challenges, including material control and accountability, and security challenges unique to MSRs;
- maintenance and operations challenges;
- program documentation challenges.

Nevertheless, the outlook for solving the above issues is positive. Multiple companies, universities, and regulatory bodies are working on their own solutions, as well as collaborative projects. International collaboration could be strengthened to facilitate MSR deployment.

Appendix I

HISTORY OF MSR TECHNOLOGY IN POLAND AND SWITZERLAND

The MSR history in Poland and Switzerland is associated to one person: Mieczyslaw Taube. He was working in the Department of Radiochemistry of the Institute of Nuclear Research in Warszawa in the 1960s. At that time, he proposed the chloride fast reactor concept [32, 33]. It was cooled by boiling aluminium trichloride, which was in direct contact with the fuel salt. By the end of the 1960s, Taube joined the Swiss Federal Institute for Reactor Research (EIR), which is the predecessor of the Paul Scherrer Institut (PSI), in Switzerland. In one of the first EIR publications, the aluminium trichloride vapour was analysed as a coolant of a fast reactor with solid fuel [34]. However, the boiling salt and the directly cooled concepts were abandoned because of many technical difficulties. In the early 1970s, a series of heterogeneous chloride fast reactor concepts were proposed by Taube [35, 405, 148], in which the chloride fissile fuel salt was typically in pins being cooled by the fertile blanket salt. These concepts were aiming at breeding in the U-Pu fuel cycle and at transmutation of legacy waste. In the case of transmuting waste, beryllium chloride was used as a diluted moderator to increase the burning rates.

When the EIR-215 report, Ref. [35], was published in 1972, Taube sent it with a letter to several other states / institutes, which were interested in MSR, including: Britain / UKAEA, France / CEA, Netherland (EURATOM) / Petten, USA / University of Tennessee and ORNL. He received many answers, such as the one from M. W. Rosenthal (ORNL) dated December 1972, which included the report ORNL-4812; however, the MSR project at ORNL was stopped soon after. At that time, three labs of EIR were partly involved in MSR neutronics, chemistry, and materials research.

The heterogeneous core layout was replaced by homogeneous concepts in the middle 1970s to avoid the technical difficulties with the separating structural material. In 1974, the chloride fast reactor concept was proposed, which combined the breeding in Th-U and U-Pu fuel cycles [36]. There is also an unpublished EIR technical memo from 1975 where the molten chloride salts were considered as an energy-storing medium [406]. In addition, a dedicated waste burner concept was proposed at that time [407]. It had a central subcritical burning zone, which was separated by a moderator from the outer driving zone, which provided neutrons to the central, or actually driven, zone. This concept is presented also in a report from 1975 [127], a summary report from 1978 [80], and in the Swiss contribution to the International Nuclear Fuel Cycle Evaluation – Working Group 8: Advanced Fuel Cycle and Reactor Concepts from 1978 [80]. In the last three named publications, E. Ottewitte is listed as EIR co-author. He published a concept operating in the Th-U cycle in 1978 [408], and defended his thesis at University of California, Los Angeles on this topic in 1982 [38]. This thesis is one of the first documents where the abbreviation MCFR (Molten Chloride Fast Reactor) is used for fast chloride-based reactors. Taube used the term MCFBR (Molten Chloride Fast Breeder Reactor) [405], and Smith, in the British report of Ref. [42], MSFR (Molten Salt Fast Reactor) since 1974. The abbreviation MSFR is, nonetheless, nowadays used rather for fluoride fast reactor. The EIR report [80] and the thesis of Ottewitte [38] include which is quite iconic for a fast MSR.

FIG. 96. Basic layouts for homogeneous fast reactors for single-fluid (left), two-fluids (middle) and three-fluids (right) and placement of fissile and fertile materials (Ref. [11]).

At the end of the 1970s, the SOFT (Salt reactor, On site reprocessing, Fast converter, Task) reactor concept was proposed by Taube. It was a homogeneous U-Pu breeder [37]. The spherical core was reflected with another chloride salt. The MSR research at EIR was almost exclusively relying on simulation at that time. One of the few small experiments was the chloride salt irradiation near the core of the EIR "Saphir" reactor (swimming-pool reactor designed by ORNL for the first Geneva conference) and chemical speciation of the produced sulphur [409]. In the 1980s, the MSR research faded out at EIR. It was restarted at PSI in 2013 with a scoping study [353] and in 2015 the breed-and-burn fuel cycle option became one of the major research directions [128][132].

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Appendix II

HISTORY OF MSR TECHNOLOGY IN CHINA

In February 1970, Premier Zhou Enlai's stated that Shanghai should develop an NPP to solve the disparity between Shanghai's economic development and energy shortage. An MSR for high temperature was initially selected for the NPP project, which would generate 25,000 KW(th) [46].

In the 1970s, a Critical Experiment Device (hereafter the device) was established for the research on the physics characteristics of MSR in the Shanghai Institute of Applied Physics (SINAP) (then Shanghai Institute of Nuclear Research), and a series of zero power experiments and related results were obtained. The device was later transformed into a Uranium-Water Lattice Experiment Device for changing the research orientation. A major advantage of this device is the flexibility of the fuel elements and moderator elements arrangement in the core allowing different moderating ratios. The purpose of the device was to experimentally verify the results from theoretical calculations; get important characteristics, such as the static and dynamic features, the unit inventory gram reactivity, and the temperature effect of the molten salt used; and obtains the relevant design data such as the control rod calibration and its temperature effect, and the fertile fuel conversion ratio [47, 48, 49, 50].

The device is an experimental, zero-power reactor moderated by graphite, fuelled with BeF_2 powder and UF_4 / ThF_4 powder mixed in proportion. The core, located in the centre, is a graphite cylinder with 497 fuel or moderator channels arranged in the shape of equilateral triangles, as shown in Figure 97 [410, 411]. As the name implies, the fuel channels are used for fuel elements insertion and the moderator channels are for graphite elements insertion for various cell structures with different moderating ratios, and the diameter of the channels is approximately 30.0 mm. The diameter and the height of the cylindrical core are 120.0 cm and 129.6 cm, respectively. The graphite reflector surrounding the core includes upper reflector which can be lift flexibly for cell replacement in the core, lower reflector, and side reflector that is octagonal in shape. The thickness of the upper and lower reflectors and the minimum thickness of the side reflector are 63.0 cm, 61.5 cm, and 63 cm, respectively. The channels in the side reflector are used for various detectors, control rods and neutron source, with a channel depth depends on its function. The air cavity between the reflector and the concrete shield can play the role of thermal insulation to some extent. The cylindrical concrete shield is in the periphery of the reactor and its thickness is 70.0 cm. In addition, the heavy shielding concrete plate above the upper reflector can be removed to satisfy the experimental requirements. A control rod was made of cadmium, and was manufactured by simply wrapping up an aluminium bar with cadmium [411]. The diameter of the cadmium rods changed with the inner aluminium bar diameter. The length of the part of the control rod wrapped with cadmium sheet was 120.0 cm, which was the same as the height of the fuel elements. For the sake of safety, there were 6 cadmium rod channels marked from No. 1 to 6 in the side reflector, as shown in Figure 97. Channels 1 and 2 were for shutdown rod insertion, channels 3 and 4 were spare shutdown rod channels, and channels 5 and 6 were for shim rod insertion and regulating rod insertion, respectively. A fuel element is made up of fuel blocks, or fuel blocks and graphite blocks alternately loaded in a graphite annular tube, as shown in the right image in Figure 97, with fuel block number and graphite block number ratios for various active region heights and fuel volumetric shares. Each fuel block contains UF_4 1.6 g and BeF_2 8.1 g, and the average fuel powder density is 0.979 g/cm^3 . The height and the diameter of a graphite block is about 39.1

mm and 20.0 mm respectively. Furthermore, there are several graphite rods inserted in the outer channels of the core with their heights almost equal to the height of the core, and they are used for adjusting the active region equivalent diameter and the graphite volume ratio [411]. The reactor could achieve a maximum power of 200 W(th), but all experiments mentioned herein were performed at zero power state and room temperature without coolant.

FIG. 97. Schematic diagrams of the Critical Experiment Device. Left and right: Horizontal and vertical cross-sections of the device, respectively. Control rod channels are marked with 1 to 6. Experimental detector and test sample location are marked with 7 to 12.

The results of the experiments performed on this device are summarized as follows:

- The critical mass calculation is consistent with the experimental value within the error range of 5%, and the physics theoretical calculation model and method are accurate.
- There is a large deviation between the measurement results of neutron flux and the calculation results.
- Molten salt at high temperature has a great effect on the corrosion of structural materials in the reactor, which impacts the safety of the reactor. The material and fuel compatibility issues have been solved, and the adoption of Hastelloy alloy will effectively reduce the corrosion, and improve the safety and extend the life of the reactor later.

In July 1973, the “Request for instructions on changing the project construction plan of the 728 Nuclear Power Plant”, explained that after two years of research and tests, many problems remained to be solved, such as dealing with high radioactivity of molten salt fuel, and finding a suitable intermediate heat carrier. Steel alloy that is resistant at high temperature and corrosion resistant did not satisfy the requirements. A chemical reprocessing process had not been found. Therefore, according to the test results, the technology of MSR required more scientific research, which would take a long time. This timing was not consistent with the original intention of the 728 project to build a NPP of medium size as soon as possible. Therefore, the MSR project was cancelled and the project 728 was replaced by pressurized water reactor test and design.

Appendix III

HISTORY OF MSR TECHNOLOGY IN FRANCE

This appendix provides detailed technical information about the French MSR designs developed between 1998 and 2008.

III.1. WORK ON GRAPHITE MODERATED MSR (MSBR TYPE): THE AMSTER CONCEPT 1998 TO 2005

The AMSTER concept was first proposed in 1999 [412] based on the Molten Salt Breeder Reactor (MSBR [103]) developed by ORNL. The MSBR was a 1 GWe reactor design based on the 8 MW(th) prototype Molten Salt Reactor Experiment (MSRE) which was successfully operated between 1965 and 1969. AMSTER used enriched uranium in this very first version, but the following studies used Thorium fuel, like the MSBR, because it allows to be near breeder in the neutron spectrum when using a graphite moderator [413, 406, 414, 415]. All technical characteristics of the MSBR were kept in AMSTER: Fluoride salt, Graphite moderator, hastelloy-n, online reprocessing of the salt (removal of the fission products (FPs)). However, the MSBR required intense reprocessing activities (whole core reprocessed in 10 days to extract ^{233}Pa as fast as possible and get a 1.06 fissile conversion ratio). Since AMSTER aimed at burning Pu and MA and not maximize breeding rate, the time for reprocessing the whole core was increased to 300 days for the soluble fission products. Two reactors were optimised in Ref. [416]: the AMSTER TRU burner, which is an actinides burner loaded with TRU elements coming from PWR used fuel, and the AMSTER iso-breeder, which is a simplified version of the MSBR, with a conversion ratio equal to one. They are briefly described next, and Table 16 presents information about their fuel.

AMSTER TRU burner. The initial loading of 82.7 t/GW(e) of Thorium and 1.16 t/GW(e) of TRU coming from PWR spent fuel represents 1.4% of the heavy nuclei. The soluble FPs are removed in 300 days, and the TRU are fully recycled within the reactor. 22 kg/TWh(e) of TRU and 76 kg/TWh(e) of ^{232}Th are consumed at equilibrium (without any need of uranium 235 addition). The ^{233}U and FPs rapidly tend to an equilibrium.

AMSTER iso-breeder. The reactor consumes only 98 kg/TWh(e) of ^{232}Th . To achieve this performance, a fertile zone around the periphery of the core was designed, representing half of the salt volume. This zone is under-moderated by increasing the diameter of the salt and thus reducing the graphite available to thermalize neutrons; see Figure 98.

TABLE 16. FUEL OF THE TWO AMSTER REACTOR TYPES WITH 300 EFPD REPROCESSING TIME ^a

	AMSTER TRU burner			AMSTER iso-breeder		
	Initial loading (kg/GWe)	Equilibrium loading (kg/GWe)	Equilibrium feeding (kg/TWhe)	Initial loading (kg/GWe)	Equilibrium loading (kg/GWe)	Equilibrium feeding (kg/TWhe)
Thorium	82 700	80 000	76	74 000	80 000	98
TRU	1 160	1 940	22		170	
²³³ U		1 240			1 460	
²³⁵ U				2 000		
²³⁸ U				8 050		
FP		750			750	

^a EFPD means effective full power day, and TWhe means Terawatt-hour electric.

FIG. 98. Representation of the two moderation zones of the AMSTER iso-breeder (black: graphite, yellow: salt).

This work has been extended to the main R&D challenges of the MSR feasibility, including materials and reprocessing simulation and safety at the French [417, 418, 419, 420, 421, 422, 423] and European levels. “The state of the art review of MSR technology, performed in the MOlten Salt reactor Technology (MOST) project [424] supported by Euratom in 2002-2004 (5th Framework Program), confirmed the potential of MSRs as breeders or burners. The main achievement of the project consists of the 9 review reports issued by the end of 2003, covering all the main aspects of MSRs, that is, reactor physics, design and operation, safety, systems and components, structural materials, fuel salt chemistry, fuel processing, economics, and proliferation resistance.”

Regarding MSR safety, codes for the simulation of transients have been developed by EDF [425] and by the partners of the MOST project [302], and applied to transients measured in the MSRE. The temperature feedback coefficients of the MSRE [426] and the MSBR [103] were re-evaluated [427]. It was shown that the Apollo2 [428] neutron transport code used by EDF was able to compute precisely the feedback coefficients measured by ORNL on the MSRE. It was concluded that the feedback coefficient computed by ORNL were underestimated, and that the MSBR feedback coefficient was slightly positive.

The MOST project concluded at the end that an MSR moderated with graphite could hardly achieve a compromise on 5 major constraints: safety (feedback coefficients), breeding, ²³³U initial inventory, moderator (graphite) longevity, and feasibility of reprocessing. The Centre National de la Recherche Scientifique (CNRS) work of optimization documented in Mathieu’s PhD thesis [429] showed that no configuration with graphite could satisfy the requirements of breeding, limited graphite swelling, and safety; see Figure 99.

FIG. 99. Main performance parameters for TMSR [59].

III.2. FROM 2004 TO 2008: WORK ON NON-MODERATED MSR

Hence, the CNRS proposed the non-moderated Thorium Molten Salt Reactor (TMSR) [59, 60, 61] in 2004, which uses a fluoride salt and is iso-breeder. See Figure 100.

FIG. 100. Neutron spectrum in different MSRs.

Practically in the same time frame, EDF defined a non-moderated iso-breeder MSR, the REBUS-3700 [62, 63] with a classical depleted U / Pu cycle. This reactor uses a chloride salt ($38\text{UCl}_3\text{-}7\text{TRUCl}_3\text{-}55\text{NaCl}$) because a preliminary study showed that $^{238}\text{U} / ^{239}\text{Pu}$ breeding is hard to reach with a fluoride salt, since the neutron moderation by fluoride is too high. This design, considered preliminary at that time, has very good safety coefficients ($-6\text{ pcm}/^\circ\text{C}$) and an initial fissile (Pu) mass comparable to other fast breeder reactors (11.9 t TRU/GWe). The initial fissile loading is 17.8 tons of TRU coming from PWR spent fuel, representing 15% of the heavy nuclei. The fuel reprocessing is then adjusted to be as slow as possible while keeping the reactor operating as a breeder (36 l/day representing 0.065% of the primary salt or a reprocessing period of 1,538 days). The fuel burnup is approximately 4.5% at equilibrium. The REBUS-3700 reactor achieves the main design goals: a breeding gain equal to 0. and a strong negative temperature coefficient of reactivity. Accordingly, REBUS-3700 is an MSR with a fast neutron spectrum and the fuel is dissolved in a chloride salt. Table 17 gives its major technical parameters.

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TABLE 17. MAJOR TECHNICAL PARAMETERS OF REBUS-3700

Parameter	Value
Technology developer, country of origin	EDF, Electricité De France (France)
Thermal/electrical capacity, MW(t)/MW(e)	3700 MW(t) / 1500 MW(e)
Moderator	None
Conversion ratio	1.03
Fuel processing	Full
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	483 kW(t)/kg
Primary circulation	Forced (13.6 m ³ /s)
Salt speed	1.2 m/s
Core inlet/exit temperatures (°C)	650 / 730
Fuel type	Chloride (38%UCl ₃ -7%TRUCl ₃ -55%NaCl)
Fuel enrichment (%)	None. It is started with 15.6% TRU from PWR and then fuelled with depleted uranium. Equivalent fissile enrichment~10%.
Fuel loading / top-up scheme	Continuous fuel make-up and fuel salt cleanup in batches with removal time of 1538 equivalent full power days (EFPD) for soluble fission products.
Main reactivity control mechanism	Large negative feedback temperature and void coefficients, and removal of fuel from the core by thermal dilation
Core diameter/height (m)	3.8 / 3.25
Core/Primary salt volume (m ³)	36.8 / 56.3
Salt density at 690 °C (g/cm ³)	3.6
Design status	Concept

At that time, the chloride salt seemed to be less known than the fluoride salt for its use as reactor fuel to researchers at EDF. However, around 2015, Terrapower, Elysium and Moltex proposed non-moderated MSR with chloride salts based on the experience of reprocessing metal fuel of the Integral Fast Reactor (IFR), that is, compatibility of chloride with a standard alloy. Hence, the Titanium vessel of REBUS could be replaced with a less expensive material like steel.

Finally, two very similar non-moderated TRU burners on inert support (without ²³⁸U or ²³²Th) were defined in Europe. The MOSART 2,400 MW(th) concept defined by Kurchatov Institute in Russia [430] (15%LiF-27%BeF₂-58%NaF dissolving trifluoride of Pu+MA produced by PWR, and lithium enriched to 99.99% ⁷Li), and the SPHINX 1,340 MW(th) project [431] defined by NRI (Czech Republic) (Fluoride salt 35%⁷LiF/38%BeF₂/27%NaF dissolving Pu+MA produced by PWR).

The European Specific Support Action named ALISIA clearly concluded in 2008 that non-moderated MSR's had great potential because there is no graphite waste and their stability due to negative feedback coefficients of reactivity [432]. The two most interesting remaining are two non-moderated concepts, one breeder (non-moderated TMSR) and one burner (MOSART). Similar behaviour in accidental transients is expected because both have a -5 pcm/ $^{\circ}$ C temperature coefficient of reactivity of the salt and the same recirculation of delayed neutron precursors.

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Appendix IV

HISTORY OF MSR TECHNOLOGY IN RUSSIA

Russia carried out R&D work on container materials, heat and mass transport properties of molten salts, irradiation of fuel salt, and related test facilities and loops were established from 1976 to 1986. Ref. [433] offers information about these activities, and in general on molten salt fuels for nuclear waste transmutation in accelerator driven systems. The following paragraphs provide additional information about some of them.

General requirements for the materials for containing fuel and coolant salts were formulated as follows:

- They must be easily produced, that is, easily subjected to treatment by pressure, and well behaved for cutting, rolling, and welding. It must also be possible to fabricate complex components of the reactor structure by using these materials.
- Pressure on walls in a typical MSR fuel circuit is less than 2 MPa. Thus, guided by this limiting value and taking a ten-fold safety factor, the candidate material must have a limiting value for lasting strength bigger than 20 MPa during 30 years of operation at a temperature of 750°C.
- The peak neutron flux and neutron fluence in the reactor vessel of an MSR are estimated as 10^{20} and 5×10^{21} n/cm² for fast (> 0.5 MeV) and thermal neutrons, respectively.
- To keep the effects of corrosion at an acceptable level (i.e., maintaining wall thickness of heat exchangers, molten salt content, and mass transfer), the depth of corrosion should not exceed 10 mm/year, and the metal should not be subjected to local corrosion (pitting or intergranular cracking).

A list of in-reactor tests regarding the radiation stability of different fluoride molten salt fuels carried out at KI is given in Several molten salt test loops operating at high temperature with forced and natural circulation were created and successfully tested in the Russian MSR program. In laboratory and in reactor molten salt test loops operated from 500 to 3500 hrs at temperatures 500-800°C working capacity of loops components and system is shown. The modes of start-up and shut down of the loops were completed, and also approaches for removing impurities and measuring redox potential were improved. In addition, the interaction with construction materials, radiation resistance, and heat and mass transfer in fluoride molten salts were studied. Table 19 presents the molten salt test loops operated by KI in the Russian Federation.

Table 18. This table presents measured values of radiation chemical yield $G(F_2)$, which is the number of F_2 molecules evolving per 100 eV of absorbed energy.

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TABLE 18. IN-REACTOR TESTS OF THE RADIATION STABILITY OF FLUORIDE MOLTEN SALT FUELS

Fuel salt (mole %)	Liquid phase		Solid phase	
	T (°C)	$G(F_2)$ (10^{-5} mol/100eV)	T (°C)	$G(F_2)$ (10^{-2} mol/100eV)
66LiF-33BeF ₂ -1UF ₄	615	7	50	1
69LiF-31BeF ₂	680	2	50	0.2
71.7LiF-16BeF ₂ -12ThF ₄ -0.3UF ₄	740	3	25	0.6
65.6 ⁷ LiF-34.1BeF ₂ -0.3UF ₄	740	0.2	25	1
73.6iF-25.9ThF ₄ -0.5UF ₄	1200	2	-	2.5
74NaF-25.5ThF ₄ -0.5UF ₄	1150	0.15	50	2

TABLE 19. LIST OF THE MOLTEN SALT TEST LOOPS OPERATED BY KI

Loop Type	Melt (% mole)	Volume (l)	T_{max} (°C)	ΔT (°C)	Operation (hrs)	Alloy
Forced convection	46.5LiF-11.5NaF-42KF	90	620	20	3,500	12H18N10T
Natural convection	92NaBF ₄ -8NaF	6	630	100	1,000	HN80MT
	72LiF-16BeF ₂ -12ThF ₄ + UF ₄	6	750	70	1,000	HN80MTY
	LiF-NaF-BeF ₂ + PuF ₃	8	700	100	1,600	Based on Ni
	LiF-NaF-BeF ₂ + Cr ₃ Te ₄	12	650	10	500	Based on Ni
Natural convection in reactor	66LiF-34BeF ₂ + UF ₄	19	630	100	500	12H18N10T
	66LiF-34BeF ₂ + UF ₄	19	750	250	750	12H18N10T

The loop SOLARIS for forced convection is operated at KI to evaluate the compatibility of stainless steel with Li,Na,K/F eutectics, and the operability of main components and systems, including centrifugal pumps, salt-air heat exchangers, and salt valves. All piping joints in contact with salt are made by argon-arc welding.

The reactor loop KUSR-2 for natural convection is being operated by KI to evaluate the behaviour of stainless steel construction material with fuel Li,Be,U/F salt at temperatures, neutron fluxes and flow rates similar to those in a typical MSR primary circuit. Investigations were carried out at the experimental reactor VVR-SM with a neutron flux up to $0.76 \cdot 10^{14}$ n/(cm²·s), and operation time of 3,500 hrs. The test section was cooled by reactor water. Heat generated due to nuclear reactions directly in the fuel salt and the thermosyphon walls is transferred from the core into the cooling zone by natural convection over the central insertion through flow distributors. In the downflow section, the heat is removed by the reactor water. In the KURS-2 loop, the possibility is provided for a preliminary fluoride passivation of gas

supply lines, a mass spectrometric analysis of gas samples, and in-line control of pressure in gas volumes.

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Appendix V

DESCRIPTION OF MSR CONCEPTS

V.1. INTRODUCTION

This appendix presents brief descriptions of MSR concepts that are currently being developed. Concepts that were proposed in the past are presented in Section 2, History of MSR Technology, and Appendices I through IV.

Practically all current MSR developers were contacted, and each was requested to provide a brief summary of its concept. This appendix presents the summaries of the concepts; unfortunately, the summaries that were not received on time for publishing are not included in this publication.

A more detailed description of MSR concepts can be found in the online IAEA's database Advanced Reactor Information System (ARIS) (if the developer of the concept provided such description to this database) [385]. This database is updated periodically. The recent IAEA publication 'Advances in Small Modular Reactor Technology Developments (2020 Edition)' [434] is a supplement to ARIS, and has a section with 10 MSRs that are SMRs.

V.2. DESCRIPTION OF MSR CONCEPTS

This subsection presents the summaries of MSR concepts. The concepts are organized according to the six major MSR families¹⁸, namely:

- I.1. Fluoride salt cooled reactors;
- I.2. Graphite-moderated MSRs;
- II.3. Homogeneous fluoride fast MSRs;
- II.4. Homogeneous chloride fast MSRs;
- III.5. Non-graphite moderated MSRs;
- III.6. Heterogeneous chloride fast MSRs.

The MSR concepts are alphabetically sorted within each family.

¹⁸ See Section 4, Classification of MSR families, for the taxonomy used in this publication.

V.2.1. Mark 1 Pebble Bed Fluoride-salt-cooled High-temperature Reactor (Mk1 PB-FHR)

Class: Graphite-based MSR;

Family: Molten salt cooled reactors (I.1);

Type: Salt cooled reactor with pebble bed fuel

TABLE 20. MK1 PB-FHR'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Solid
Reactor Spectrum	Thermal
Salt Type	Fluorides
Coolant	Salt
Actinide	U-Pu

TABLE 21. MK1 PB-FHR'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	University of California, Berkeley, USA
Thermal/electrical capacity, MW(t)/MW(e)	236 / 100
Moderator	Graphite and salt
Conversion ratio	Less than 1
Fuel processing	Once through
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	1869
Primary circulation	Forced circulation
Primary pressure (MPa)	0.3
Core inlet/exit temperatures (°C)	600 / 700
Fuel type	TRISO with enriched uranium kernel
Fuel enrichment (%)	19.8
Fuel loading / top-up scheme	Online refuelling
Main reactivity control mechanism	Control rods
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m ²)	60
Reactor vessel's height/diameter (m)	2.5/2.1
Seismic design	Base isolation

Table 22. MK1 PB-FHR'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Distinguishing features	Pebbles are inserted from the bottom of the core and extracted at the top
Design status	Pre-conceptual design

The Mk1 PB-FHR uses solid fuel made of coated particles cooled by the fluoride salt FLiBe (${}^7\text{Li}_2\text{BeF}_4$). Fluoride salt coolants have uniquely high volumetric heat capacity, low chemical reactivity with air and water, very low volatility at high temperature, effective natural circulation heat transfer, and high retention of most fission products. The Mk1 PB-FHR is an SMR that does not use an intermediate coolant loop and instead directly heats the power conversion fluid. It eliminates the conventional reactor guard vessel used in sodium fast reactors and instead uses a refractory reactor cavity liner system. All components for the Mk1 design are modular in their construction and are transportable by rail. The reference configuration for the Mk1 site is based on 12 Mk1 units that are capable of producing 1,200 MW(e) of base load electricity and ramping to a peak power output of 2,900 MW(e) through the addition of burning natural gas or other combustible fluid as a topping cycle in the gas turbine. The core incorporates 3-cm diameter spherical pebble fuel elements with coated uranium particles in an annular fuel zone and a low-density graphite core. One Mk1 pebble contains 1.5 g of uranium enriched in U-235 to 19.9%, which is encapsulated inside 4,730 coated particles. The very low circulating power for the coolant in salt-cooled reactors, compared to helium-cooled reactors, makes it practical to use 3-cm pebbles. This small-pebble design doubles the pebble surface area per unit volume and halves the thermal diffusion length, enabling a substantial increase in power density while maintaining relatively low peak fuel particle temperature. Low fuel temperature reduces the thermal transient caused by hypothetical anticipated transient without scram (ATWS) accidents.

The design uses a buoyant control rod system for normal reactivity control, and it also provides a passive shutdown function because the buoyant rods will insert if the reactor coolant temperature in the control rod channel exceeds 615°C, the buoyant stability limit. If electrical power is interrupted to the drive mechanisms for the motors of the control rod and shutdown blade cable drums, the control rods will insert and shut the reactor down. The insertion of the shutdown blades therefore provides a reserve shutdown function. The safety function of emergency decay heat removal is also controlled by passive mechanisms. The design employs a passive check valve to activate heat transport driven by natural circulation from the primary coolant to a set of three Direct Reactor Auxiliary Cooling System (DRACS) loops and ultimately to Thermosyphon-cooled Heat Exchangers (TCHXs) upon Loss of Flow Condition (see Figure 101). In addition to the passive emergency decay heat removal provided by the DRACS, the power conversion system and the normal shutdown cooling system provide heat removal capability and defense in depth in assuring adequate core heat removal.

V.2.2. FUJI

Class: Graphite-based MSR;
 Family: Graphite moderated MSR (I.2);
 Type: Single-fluid Th-U breeder

TABLE 23. FUJI'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Thermal
Salt Type	Fluorides
Coolant	Fuel Salt
Actinide	U-Th, or Pu-Th

TABLE 24. FUJI'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	International Thorium Molten-Salt Forum, Japan
Thermal/electrical capacity, MW(t)/MW(e)	450 / 200
Moderator	Graphite
Conversion ratio	Equal to 1
Fuel processing	Off-site reprocessing
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	400
Primary circulation	Forced circulation
Primary pressure (MPa)	0.5 (by pump head)
Core inlet/exit temperatures (°C)	565 / 704
Fuel type	Molten salt with Th and U
Fuel enrichment (%)	2.0 (0.24% ²³³ U + 12.0%Th) Pu or LEU can be used.
Fuel loading / top-up scheme	Slight adjustment at every month
Main reactivity control mechanism	Control rod, or pump speed, or fuel concentration
Approach to engineered safety systems	Passive safety
Design life (years)	30
Plant footprint (m ²)	< 5,000
Reactor vessel's height/diameter (m)	5.4 / 5.34 (inner)
Seismic design	0.6g to 2.0g, depending on the reactor location

TABLE 24. FUJI'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Distinguishing features	High safety, high economic performance, contribution to non-proliferation, and fuel cycle flexibility
Design status	3 experimental MSR's were built [319]. Detailed design will be started within 2-3 years.

The typical electric output of MSR-FUJI is 200 MWe with 44% thermal efficiency. Further, FUJI's power is flexible from 100 MWe to 1,000 MWe [319]. In addition to the above high thermal efficiency, the low manufacturing cost owing to the simple core structure and high fuel efficiency owing to the high conversion ratio contribute to high economic performance. FUJI can consume Pu as the fissile material, and can contribute to reduce the proliferation risk caused by Pu from LWR spent fuel. It can also be used to transmute long-lived minor actinides to shorter ones. Reactor vessel, pumps, and heat exchangers are enclosed in the high temperature containment. FUJI has very favourable safety characteristics that essentially exclude the possibility of severe accidents. It is expected that FUJI can be deployed in less than 10 years [318]. FUJI is based on the results obtained by ORNL in the 1960s, and has been optimized as a small sized plant and further simplified by removing the online reprocessing facility. Based on the operating experience at three experimental MSR's in ORNL, it has been verified that FUJI is feasible. The steam generator is a major unverified component, but it may be developed based on experience from the fast breeder reactor (FBR) and the recent supercritical power station technology.

FUJI adopts a passive safety system to improve the plant's safety, as well as the economics. Molten fuel salt can be drained to a sub-critical drain tank through a freeze valve. Since gaseous fission products are always removed from molten fuel salt, the risk of releasing such products during postulated accidents is reduced. FUJI is operated at very low pressure, and a thick reactor vessel and pipes are not required. There are no fuel assemblies or complex core internal structure, with the only component within the reactor vessel being the graphite moderator. Based on these design principles, in-factory fabrication would be simple. Figure 102 presents an aerial view of MSR-FUJI.

FIG. 102. Aerial view of MSR-FUJI.

V.2.3. Integral Molten Salt Reactor (IMSR)

Class: Graphite-based MSR;

Family: Graphite moderated MSR (I.2);

Type: Uranium converters and other concepts

TABLE 25. IMSR'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Thermal
Salt Type	Fluorides
Coolant	Fuel Salt
Actinide	U-Pu

TABLE 26. IMSR'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	Terrestrial Energy Inc., Canada
Thermal/electrical capacity, MW(t)/MW(e)	~440MW(t)/~195MW(e)
Moderator	Graphite
Conversion ratio	Less than 1
Fuel processing	Once through
Kilowatt per total fissile load (kW/kg)	Not provided
Primary circulation	Forced
Primary pressure (MPa)	Hydrostatic, near atmospheric
Core inlet/exit temperatures (°C)	~620 / ~700 (°C)
Fuel type	UF ₄ standard assay LEU
Fuel enrichment (%)	~2% Startup, 4.95% Makeup Fuel
Fuel loading / top-up scheme	Initial ~2% beginning-of-life fuel load with periodic makeup fuel additions over 7-year Core-unit lifetime. No fuel removal during lifetime of 7 years.
Main reactivity control mechanism	Negative reactivity coefficient of temperature and shutdown rods
Approach to engineered safety systems	Passive safety systems
Design life (years)	7 years per Core-unit, 56 years for plant
Plant footprint (m ²)	45,000
Reactor vessel's height/diameter (m)	~10m/3.7m

TABLE 26. IMSR'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Seismic design	0.3g for Eastern North American siting
Distinguishing features	Passive safety approach, integral primary systems within a 7-year replaceable Core-unit
Design status	Basic engineering

The Integral Molten Salt Reactor (IMSR) is an advanced, 440 Megawatts-thermal, liquid-fuelled, small modular, molten salt reactor (IMSR400). It features a sealed integral Core-unit housing the liquid fuel, moderator, pumps, primary heat exchangers and shutdown rods. The entire Core-unit is replaced at the end of its 7-year life allowing factory production levels of quality control and economy while avoiding opening the reactor vessel during its lifetime. A non-fuelled liquid fluoride salt loop transfers heat from the Core-unit to a third salt loop which, utilizing solar nitrate salts, transfers the heat to a separate building where it generates superheated steam for electric power generation or is used for industrial process heat applications. Figure 103 depicts the IMSR Core-unit and its major components shown in an operating silo.

FIG. 103. IMSR Core-unit and its major components shown in an operating silo.

Utilizing liquid fuel, the IMSR realizes a naturally strong negative reactivity coefficient of temperature. This desirable safety characteristic provides a self-governing, stable temperature regime, and establishes an inherently safe operating profile in which reactor power is inherently controlled to demanded power in the short term without the need for manipulation of any reactivity control device. In the long term, reactivity is controlled by small, routine manual additions of fuel salt. There are no restrictive neutron flux limits compared to limits imposed upon traditional water-cooled reactors related to fuel cladding integrity. Hence, neutron flux can transiently increase by a significant amount without any negative effects on core integrity. For these reasons, the IMSR does not require any control rods nor automatic flux control algorithms for reactor power control. The rapid response to temperature changes, and low fission product poison density also enables load-following capability facilitating backing-up variable wind and solar power generation; a capability which supports electric grid reliability and wind and solar deployments.

In summary, the liquid fuel IMSR allows significant simplification of the power plant design which eases construction and commissioning capital cost burdens and improves overall operating cost performance.

V.2.4. *smTMSR-400*

Class: Graphite-based MSR;
 Family: Graphite moderated MSRs (I.2);
 Type: Uranium converters and other concepts

TABLE 27. IMSR'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Thermal
Salt Type	Fluorides
Coolant	Fuel Salt
Actinide	U-Th

TABLE 28. IMSR'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	SINAP, CAS, China
Thermal/electrical capacity, MW(t)/MW(e)	400 / 168
Moderator	Graphite
Conversion ratio	~0.6
Fuel processing	Limited
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	~1000kW/kg
Primary circulation	Forced circulation
Primary pressure (MPa)	0.5 MPa
Core inlet/exit temperatures (°C)	650 / 700
Fuel type	LiF-BeF ₂ -UF ₄ -ThF ₄
Fuel enrichment (%)	19.75wt%
Fuel loading / top-up scheme	Th+U loading initial. LEU addition online. FP gas removal online. Batch reprocessing offline
Main reactivity control mechanism	Control rods. Negative reactivity feedback. Online fuel addition. Draining off fuel salt
Approach to engineered safety systems	Passive
Design life (years)	60
Plant footprint (m ²)	-
Reactor vessel's height/diameter (m)	~10 / 3.88
Seismic design	0.3g

TABLE 28. IMSR'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Distinguishing features	Replaceable reactor pressure vessel (8-10 years). Passive safety. Greater than 40% power contributed by thorium.
Design status	Pre-conceptual design

The smTMSR-400 is a 400MW(th) / 168 MW(e) small modular Thorium Molten Salt Demonstration Reactor. It is designed as a thorium convertor and in situ burner driven by low enriched uranium and will be applied as a heat source at high temperature, which not only can be used for electricity generation, but also can satisfy diversified energy demands. The smTMSR-400 consists of reactor module, heat transfer system, heat storage system, heat utilization system and other auxiliary systems, as shown in Figure 104.

FIG. 104. Diagram of smTMSR-400.

The reactor module is designed as a compact loop structure with the reactor core, three plate salt-salt heat exchangers, one centrifugal pump, and connecting pipelines. This module is inside a safety vessel that provides an additional radioactive confinement barrier. The diameter of the reactor module is about 3.8 m, so it is suitable for railway transportation. The heat transfer system is designed to transfer the heat to the heat storage system, and to isolate a radioactive release. The secondary coolant is NaF-BeF₂ due to its good thermal properties, good chemical stability, and compatibility with the fuel salt. The operation temperature of this coolant is from 600°C to 680°C. The heat storage system is in standby mode and its design employs a double tank structure for meeting demands for electricity during peak times and heat recovery. A chlorine salt of solar salt will be used as coolant and storage media. The temperatures of the cold and hot tanks are 290°C and 650°C, respectively. The heat utilization system can be designed for electricity generation by using a helium/air/CO₂ Brayton power cycle, and for other non-electric applications, such as seawater desalination, heat supply, and hydrogen production.

Other auxiliary systems include fuel management system, online fission gas and tritium removal system, and two passive decay heat removal systems (DHRs). The fuel management system is used for initial fuel loading, online fuel addition, and fuel salt discharge into drain tank under normal and accident conditions. Fission gases and some noble metals are removed from the reactor module by entrained cover gas. Then they flow into a fission gas removal system for decay and separation. If the heat removal used during normal operation fails, two kinds of passive DHRs will provide cooling of fuel salt for the long term. One is located around the safety vessel in the silo (DHRs1), and the other is in the fuel salt drain tank below the reactor vessel (DHRs2).

V.2.5. ThorCon

Class: Graphite-based MSR;
 Family: Graphite moderated MSR (I.2);
 type: Uranium converters and other concepts

TABLE 29. THORCON'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Thermal
Salt Type	Fluorides
Coolant	Fuel Salt
Actinide	U-Th

TABLE 30. THORCON'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	ThorCon US, USA
Thermal/electrical capacity, MW(t)/MW(e)	557 MW(t) / 258 MW(e)
Moderator	Graphite
Conversion ratio	Less than 1
Fuel processing	Limited, future uranium re-enrichment
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	1,000
Primary circulation	Forced
Primary pressure (MPa)	1.2
Core inlet/exit temperatures (°C)	565 / 704
Fuel type	NaF-BeF ₂ -ThF ₄ -UF ₄ 76/12/10.2/1.8 mol %
Fuel enrichment (%)	19.85%
Fuel loading / top-up scheme	High-assay, low-enriched uranium (HALEU) makeup fuel from tank within Can
Main reactivity control mechanism	Negative temp coefficient, salt flow rate; fissile/fertile additions
Approach to engineered safety systems	Intrinsic, passive, using natural circulation, water evaporation
Design life (years)	Can life: 4, Plant life: 80
Plant footprint (m ²)	20,000 for 500 MW(e)
Reactor vessel's height/diameter (m)	5.590 / 4.916

TABLE 30. THORCON'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Seismic design	Hull on shear-limited sand; reactor-containing Can on elastomeric bearings, stabilized by dashpots
Distinguishing features	Low cost, full passive safety, short construction time
Design status	Complete basic design

ThorCon is a nuclear power plant in a hull. The plant uses an MSR, and the fuel is in molten salt circulated by a pump and passively drained in the event of an accident. The reactor operates at garden hose pressures using normal pipe thicknesses and it is easily automated. The plant is ship-style, with steel-plate construction. Cooling ponds passively remove decay heat. From each power module, water naturally circulates from a cold wall around the Can containing the reactor (Pot). Intermediate molten salt loops transfer heat to the steam generator. Basement water below the steam generator provides a third, passive, long-term decay heat sink. The balance of the plant is similar to that of a super-critical coal-fired power plant. The yellow rectangles in the figure are hatches for access by the cranes. Figures 105 and 106 present the plant, and the Can and Pot, respectively.

FIG. 105. ThorCon plant.

FIG. 106. ThorCon's Pot in Can in Cold Wall.

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V.2.6. Molten Salt Actinide Recycler & Transformer System with and without Th-U Support (MOSART)

Class: Homogeneous MSR;
 Family: Homogeneous fluoride fast MSR (II.3);
 Type: Pu containing fluoride fast reactor

TABLE 31. MOSART'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Fast / Epithermal
Salt Type	Li,Be,An/F; An means Actinide
Coolant	Fuel Salt
Actinide	TRU-Th-U

TABLE 32. MOSART'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	ROSATOM/NRC-Kurchatov Institute, the Russian Federation
Thermal/electrical capacity, MW(t)/MW(e)	2,400 / 1,000
Moderator	None
Conversion ratio	Up to 1
Fuel processing	Full
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	68.5 (per kg of TRU)
Primary circulation	Forced
Primary pressure (MPa)	1.5
Core inlet/exit temperatures (°C)	630 / 750
Fuel type	TRU-Th-U fluorides
Fuel enrichment (%)	MA to TRU ratio (MA/TRU) \approx 0.1 (²³⁸ Pu: 3.18%; ²³⁹ Pu: 43.93%; ²⁴⁰ Pu: 21.27%; ²⁴¹ Pu: 13.52%; ²⁴² Pu: 7.88%; ²³⁷ Np: 6.42%; ²⁴¹ Am: 0.55%; ²⁴³ Am: 2.33%)
Fuel loading / top-up scheme	Continuous fuel make-up and fuel salt cleanup in batches with removal time of 300 EFPD for soluble fission products
Main reactivity control mechanism	Fuel salt make-up and control rods
Approach to engineered safety systems	Inherent & Passive Safety

TABLE 32. MOSART'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Design life (years)	50
Plant footprint (m ²)	-
Reactor vessel's height/diameter (m)	11.2 / 4.5
Seismic design	Seismic capacity for SL2 level is 7 (in MSK scale number)
Distinguishing features	Flexibility of the fuel cycle, significant fuel quantities outside the core
Design status	Conceptual design

MOSART system represents a concept of nuclear power reactors using a molten $73\text{LiF}-27\text{BeF}_2\text{-AnF}_n$ salt mixture (in mole %) in the primary circuit. Molten salt serves as both fuel and primary coolant. The secondary coolant salt does not contain fissile or fertile materials. The core would generate about 2.4 GW(t) at conditions affording efficient transmutation and recycling of TRUs from uranium and mixed oxide fuels (UOX and MOX) from VVER-1000 SNF. The reference core uses a fast neutron spectrum, so it does not have any moderator. Radial, bottom, and top reflectors are attached to the reactor vessel. This leaves a gap filled with fuel salt surrounding the core to cool the bottom, radial and top reflectors, and reactor vessel. Cooling of the upper and lower reflectors is achieved by circulating fuel salt through openings in the lower and upper collectors, respectively. In nominal conditions, the fuel salt passes over the bottom reflector and enters the core at 630°C . The core outlet temperature increases up to 750°C , and the primary salt transfers the heat from the core to the secondary salt in the primary heat exchanger. The fuel salt mixture is circulated through the core by four pumps operating in parallel. This design is being studied in different configurations, which consider core configurations with single and two fluids, with and without Th-U support for startup loadings and make-up scenarios with different MA/TRU ratios, as well as addition of uranium after SNF reprocessing. This system can serve as an efficient burner of TRU elements from SNF, and has a breeding capability when using ^{232}Th as a fertile addition. Figure 107 shows the fuel circuit.

FIG. 107. MOSART fuel circuit.

V.2.7. Molten Salt Fast Reactor (MSFR)

Class: Homogeneous MSR;

Family: Homogeneous fluoride fast MSR (II.3) (optionally: Homogeneous chloride fast MSR (II.4));

Type: Fluoride fast Th-U breeder (optionally: Chloride fast U-Pu breeder)

TABLE 33. MSFR'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Fast
Salt Type	Fluorides (LiF-(HN)F ₄) or Chlorides (NaCl-(HN)Cl ₃)
Coolant	Fuel Salt
Actinide	U-Pu, U-Th, TRU-Th, or TRU

TABLE 34. MSFR'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	CNRS, France
Thermal/electrical capacity, MW(t)/MW(e)	3,000 MW(t)
Moderator	None
Conversion ratio	Greater than 1
Fuel processing	Full
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	400 kW(t)/kg
Primary circulation	Forced (5 m ³ /s)
Primary pressure (MPa)	< 0.5 MPa
Core inlet/exit temperatures (°C)	650 / 750
Fuel type	Fluoride (LiF-(HN)F ₄)
Fuel enrichment (%)	None (U-233), or 13% if started with TRU
Fuel loading / top-up scheme	18 m ³ – Fuelling and fuel processing during reactor operation
Main reactivity control mechanism	Large negative feedback temperature and void coefficients, and removal of fuel from the core by thermal dilation
Approach to engineered safety systems	Combination of passive and active systems
Design life (years)	60 years - 10 years (internals)

TABLE 34. MSFR'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Plant footprint (m ²)	-
Reactor vessel's height/diameter (m)	6m / 6m
Seismic design	The compactness of the fuel circuit inside the core vessel leads to seismic robustness
Distinguishing features	Fuel composition can be changed during operation. Passive cooling after shutdown.
Design status	Concept (preliminary design for safety analysis leading to safety-by-design)

The MSFR is a homogeneous reactor concept using a circulating fuel salt also used as coolant, with a core that contains no moderator and exhibits fast neutron spectrum. The circulation of the fuel salt in pipes is avoided thanks to an integrated geometry of the fuel circuit, where heat exchangers and pumps are integrated in the core vessel. Heat is extracted from the core by an intermediate molten salt. In case of failure of the cooling systems (fuel or coolant circuits), the fuel is passively drained into a dedicated emergency draining system (EDS) where it is passively cooled by natural gas or air convection. Volatile species are continuously extracted by gas bubbling and stored and processed in specific gas casings inside the fuel casing. No volatile fluid is present in the reactor and thus depressurization of gas-containing casings would only release a very limited amount of gases in the reactor building. Figure 108 is a schematic layout of the MSFR fuel circuit.

FIG. 108. Schematic layout of the MSFR fuel circuit.

The MSFR has low core reactivity due to the fuelling and fuel processing during reactor operation. Control rods or neutron poisons are thus not necessary, also due to the large negative feedback temperature and void coefficients which allow a reactivity control based on the balance between the power generated in the fuel salt and the power extracted in the heat exchangers. The absence of rods for absorbing neutrons simplifies the reactor operation and eliminates some accident initiators (e.g., ejection of a control rod).

The "reference MSFR" configuration produces 3 GW(th) and uses a fluoride salt, has been studied for more than 15 years at CNRS and in European projects, and is mainly adapted to the Th-U cycle in breeder mode. In this configuration, the fuel salt is based on the LiF-ThF₄ eutectic and contains UF₄ and PuF₃ as fissile salts. Li is enriched in ⁷Li to limit tritium production. ²³³U is bred from ²³²Th in the fuel salt and in a separate LiF-ThF₄ blanket salt (breeder version). The fissile content in the fuel salt can be adjusted without stopping the reactor by liquid salt transfer. The fuel and the blanket salts are processed on-site at a rate of about 10 litres per day and per GW(th), for limiting the fission product concentration in the fuel and extracting the fissile matter produced in the blanket.

V.2.8. Compact Molten Salt Reactor (CMSR)

Class: Heterogeneous MSR;

Family: Non-Graphite Moderated MSR (III.5);

Type: Liquid moderator heterogeneous MSR

TABLE 35. CMSR'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Thermal
Salt Type	Fluoride
Coolant	Fuel Salt and Other
Actinide	U (or other cycle)

TABLE 36. CMSR'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	Seaborg Technologies, Denmark
Thermal/electrical capacity, MW(t)/MW(e)	250/100
Moderator	NaOH
Conversion ratio	-
Fuel processing	Once through
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	200
Primary circulation	Forced
Primary pressure (MPa)	<1
Core inlet/exit temperatures (°C)	600/700
Fuel type	Fluoride salt
Fuel enrichment (%)	HALEU (as per IAEA definition)
Fuel loading / top-up scheme	Batchwise, 12 year fuel cycle
Main reactivity control mechanism	Control rods
Approach to engineered safety systems	Passive and active
Design life (years)	12 (reactor module), 24 (plant), 36 (plant life extension)
Plant footprint (m ²)	98,4m x 32m (LxB) for a 200MWe plant
Reactor vessel's height/diameter (m)	2.4/2.2
Seismic design	N/A

TABLE 36. CMSR'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Distinguishing features	No refuelling or reprocessing, liquid moderator
Design status	Basic design, first stage of regulatory approval completed

The CMSR is an advanced, small and modular, MSR with thermal neutron spectrum. It is characterised by the use of a proprietary alkali-hydroxide moderator salt. The moderator salt, containing hydrogen, allows for a compact form factor that simplifies central manufacturing and decommissioning of the complete power plant, and enables simple transport to and from the site where power is produced. The liquid moderator also does not accumulate irradiation damage, enabling a long reactor service lifetime of 12 years which is also the plant refuelling interval. One unit produces up to 250 MW(th) in the form of superheated steam, which can be used as industrial heat, or to produce 100 MW(e), or for both purposes. The reactor does not require, nor does it allow, any access to the fuel salt during the service lifetime of 12 years. A conceptual illustration of the CMSR is provided in Figure 109. A guiding principle in the development of the CMSR design is pragmatism. Consequently, the use of established solutions and components is preferred to the extent possible to facilitate rapid deployment and scale-up to importantly displace fossil fuels in time frames relevant for global warming. Inherent safety is provided by passive means, supplemented with active reactor control and protection systems, in accordance with the principles of defence in depth.

Multiple CMSRs will be installed together in floating Reactor Barges that are assembled and tested in a shipyard. The Reactor Barge is of modular build and can hold between 2–10 CMSRs for a total thermal power capacity of 500–2500 MW, corresponding to 200–1000 MW(e). Upon completion and testing, a Reactor Barge is towed to the production site and moored in sheltered waters. A Reactor Barge has a nominal service lifetime of 24 years, during which it needs to be refuelled once.

FIG. 109. Concept illustration of the CMSR primary circuit and fuel salt drain tank.

V.2.9. Copenhagen Atomics Waste Burner

Class: Heterogeneous MSR;

Family: Non-Graphite Moderated MSR (III.5);

Type: Solid moderator heterogeneous MSR

TABLE 37. COPENHAGEN ATOMICS WASTE BURNER'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Thermal
Salt Type	Fluorides
Coolant	Fuel Salt
Actinide	U-Th, TRU-Th

TABLE 38. COPENHAGEN ATOMICS WASTE BURNER'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	Copenhagen Atomics, Denmark
Thermal/electrical capacity, MW(t)/MW(e)	100 MW(t)
Moderator	Heavy water / ⁷ LiOD (lithium-7 deuterioxide)
Conversion ratio	Greater than 1 (when fuel composition has converged)
Fuel processing	Mechanical processing
Kilowatt per total fissile load (kW / kg)	
Primary circulation	Forced circulation
Primary pressure (MPa)	0.05 - 0.25
Core inlet/exit temperatures (°C)	600 / 650-700
Fuel type	LiF-ThF ₄ -PuF ₄
Fuel enrichment (%)	None
Fuel loading / top-up scheme	None
Main reactivity control mechanism	Level of liquid moderator
Approach to engineered safety systems	Passive
Design life (years)	5 - 10 years
Plant footprint (m ²)	
Reactor vessel's height/diameter (m)	12 / 2.5
Seismic design	

TABLE 38. COPENHAGEN ATOMICS WASTE BURNER'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Distinguishing features	Liquid moderator, low fissile inventory, active electromagnetic bearing canned pumps.
Design status	Conceptual

The Copenhagen Atomics waste burner is a heavy water moderated, single fluid, fluoride salt based, thermal spectrum, and autonomously controlled MSR. The core, fission product extraction and separation systems, dump tank, primary heat exchanger, pumps, valves, and compressors are all contained in a leak tight stainless steel containment, the size of a 40 foot shipping container. Figure 110 presents this reactor concept.

FIG. 110. Copenhagen Atomics Waste Burner.

The heavy water is unpressurized, thermally insulated from the salt, and continuously drained and cooled to 50°C. A molten lithium-7 deuterioxide (${}^7\text{LiOD}$) moderator variant is also being researched as a liquid moderator. Both moderator options allow for an excellent neutron economy and low fissile inventory. The liquid moderator level is used as the main reactivity control mechanism and to compensate for fluctuations in reactivity due to fuel composition changes.

Active development is focused on the test and validation of materials, flanges, valves, pumps, heat exchangers, salt purification, chemistry control systems, measurement systems, and autonomous control systems. All components are designed to last for the lifetime of the reactor (5 - 10 years) without any service; for example, this is achieved for the pumps by using active electromagnetic bearings and a canned pump design.

Mechanical separation of fission products is achieved through vacuum spraying, where the salt leaving the core is sprayed into a chamber which is at a partial vacuum pressure, and the volatility of intermediate decay products is used to extract a large fraction of the fission products. The extracted fission products are separated and stored in tanks inside the leak tight stainless steel containment.

During early use of the reactor, starting with the initial $\text{LiF-ThF}_4\text{-PuF}_4$ fuel composition, the ${}^{233}\text{U}$ production benefits from the high number of excess neutrons from the fission of plutonium. Later, the breeding process profits from the superior neutron economy of ${}^{233}\text{U}$ as the fuel converges towards its equilibrium composition of $\text{LiF-ThF}_4\text{-}{}^{233}\text{UF}_4$. Thus, when the fuel has reached this composition, the Copenhagen Atomics Waste Burner is expected to be a breeder reactor.

V.2.10 Moltex SSR-W300

Class: Heterogeneous MSR; Family: Heterogeneous Chloride Fast MSR (III.6); Type: Heterogeneous salt cooled fast MSR

TABLE 39. SSR-W300'S MAIN CHARACTERISTICS

Characteristic	Value
Fuel State	Liquid
Reactor Spectrum	Fast
Salt Type	Chlorides
Coolant	Other
Actinide	TRU

TABLE 40. SSR-W300'S MAJOR TECHNICAL PARAMETERS

Parameter	Value
Technology developer, country of origin	Moltex Energy, UK and Canada
Thermal/electrical capacity, MW(t)/MW(e)	750/300 baseload, 750/900 peak
Moderator	None
Conversion ratio	Less than 1
Fuel processing	Full
Kilowatt per total fissile load (kW-thermal / kg of fissile material)	~100
Primary circulation	Forced
Primary pressure (MPa)	~0.1
Core inlet/exit temperatures (°C)	525/590
Fuel type	Molten chloride salt fuel
Fuel enrichment (%)	Reactor grade plutonium
Fuel loading / top-up scheme	Refuelling at power
Main reactivity control mechanism	Regulation through fuel temperature coefficients; refuelling mechanism; shutdown with boron carbide assemblies
Approach to engineered safety systems	Inherent safety features, dedicated passive safety systems
Design life (years)	60
Plant footprint (m ²)	22,500
Reactor vessel's height/diameter (m)	10/6
Seismic design	0.3g peak ground acceleration

TABLE 40. SSR-W300'S MAJOR TECHNICAL PARAMETERS (cont.)

Parameter	Value
Distinguishing features	Molten salt fuel in conventional fuel assemblies; burning of nuclear waste; thermal energy storage to allow operation as peaking plant; very low cost
Design status	In transition from conceptual to engineering; Canadian vendor design review in progress

The Stable Salt Reactor Wasteburner (SSR-W300) is unique in its use of molten salt fuel replacing solid pellets in conventional fuel assemblies. This brings the major advantages of safe molten salts without the technical hurdles of managing a mobile liquid fuel. The reactor is fuelled with very low purity, reactor grade plutonium recycled from stocks of spent uranium oxide fuel and produced by a process called WATSS (Waste to Stable Salt), which is low cost. The reactor outputs its heat as a stream of molten nitrate salts which can be stored in large volume at low cost, making the SSR-W300 a peaking power plant at low cost, rather than being restricted to baseload operation. The steam cycle is identical to that in a traditional plant using fossil fuel, and it can be operated completely independently of the nuclear plant. Figure 111 presents a sectional view of this reactor concept.

FIG. 111. A sectional view of the SSR-W300.

The fuel is a molten chloride salt fuel within vented fuel tubes; the tubes are arranged in a conventional hexagonal array. The primary coolant of the reactor is a molten ZrF_4 -KF salt. The secondary coolant of the reactor is a molten nitrate salt. The SSR-W300 is designed for countries with significant stocks of spent nuclear fuel. The reactor burns the full higher actinide component of that fuel leaving a waste stream which contains only fission products that are relatively short lived. The fuel cost is expected to be negative, net of the reduced liability cost for disposal of the original spent fuel. The SSR-W300 is designed to generate peaking electrical power economically and efficiently, but with the reactor itself running at constant power. It therefore fills the need in national power systems for low-carbon flexible generation to complement intermittent renewable energy sources.

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ACRONYMS AND ABBREVIATIONS

2D	two dimensional
3D	three dimensional
ACSEPT	Actinide recycling by separation and transmutation
ADS	accelerator driven system
AEC	(US) Atomic Energy Commission
AECL	Atomic Energy of Canada Limited
AGR	(UK) advanced gas-cooled reactor
AHTR	advanced high temperature reactor
ALISIA	assessment of liquid salts for innovative applications (project)
AMSB	accelerator molten-salt breeder
AMSTER	actinide molten salt transmuter
ANP	aircraft nuclear propulsion
An(s)	actinide(s)
ANS	American Nuclear Society
APPM	atomic parts per million
ARE	aircraft reactor experiment
ARIS	advanced reactor information system (IAEA database)
ARPA-E	(US) Advanced Research Projects Agency – Energy
ART	aircraft reactor test
ASME	American Society of Mechanical Engineers
ASTRID	advanced sodium technological reactor for industrial demonstration
ATHESC	Advanced characterization and thermodynamic study of CRUD
ATWS	anticipated transient without scram
B&B	breed-and-burn
BARC	Bhabha Atomic Research Center (India)
BDBA	beyond design basis accident
BPVC	(ASME) boiler & pressure vessel code
BWR	boiling water reactor
CALPHAD	calculation of phase diagrams (code)
CANDU	Canadian deuterium uranium
CAS	Chinese Academy of Sciences
CEA	Atomic Energy Commission (Commissariat à l'Energie Atomique) (France)
CEMHTI	conditions extrêmes et matériaux: haute température et irradiation (France)
CENDL	Chinese evaluated nuclear data library
CFD	computational fluid dynamics
CFRE	circulation fuel reactor experiment
CHX	conversion heat exchanger

CIET	compact integral effects test (UC Berkeley)
CIRTEN	Consorzio Interuniversitario per la Ricerca TEcnologica Nucleare (Intra-University Consortium for Nuclear Technology Research) (Italy)
CMSR	compact molten salt reactor
CNER	Centre for Nuclear Energy Research (Canada)
CNL	Canadian Nuclear Laboratories
CNRS	Centre National de la Recherche Scientifique (France)
CNSC	Canadian Nuclear Safety Commission
CORYS	COmpagnie de Réalisation Industrielle de Simulateurs
CR	conversion ratio
CRL	(CNL's) Chalk River Laboratories
CRP	(IAEA) coordinated research project
CV	cyclic voltammogram
DAES	development of advanced engineering solutions
DBAs	design basis accidents
DFR	dual fluid reactor
DGFEM	discontinuous Galerkin finite elements
DHR	decay heat removal
DHRS	decay heat removal system
DHX	DRACS heat exchanger
DMSP	Delft molten salt project
DMSR	denaturated molten salt reactor
DNP	delayed neutron precursor
DOE	(US) Department of Energy
DPA	displacements per atom
DRACS	direct reactor auxiliary cooling system
DSC	differential scanning calorimetry
DTU	Technical University of Denmark
DYNASTY	dynamics of natural circulation for molten salt internally heated
EDC	Experimental Demonstration Centre (Russia)
EDF	Electricité de France
EDS	emergency draining system
EFPD	equivalent full power days
EFPY	equivalent full power year
EIR	Eidgenössisches Institut für Reaktorforschung (Switzerland)
ENDF	evaluated nuclear data file
EPFL	École polytechnique fédérale de Lausanne (Switzerland)
EQL	equilibrium
ETH	Eidgenössische Technische Hochschule (Zürich)

EU	European Union
EURATOM	EUROpean ATOMIC energy community
EVOL	evaluation and viability of liquid (fuel reactors) (project)
EXAFS	extended x-ray absorption fine structure
FFFER	forced fluoride flow for experimental research
FFMEA	functional failure modes and effects analysis
FHR	fluoride salt-cooled high-temperature reactor
FIHMA	fission of initial heavy metal atoms
FLiBe	LiF – BeF ₂ mixture (2LiF-BeF ₂)
FLINAK	LiF-NaF-KF mixture
FMSR	fast molten salt reactor
FOAM	field operation and manipulation (see OF)
FP	fission product
FP7	(EURATOM-ROSATOM collaborative project) framework program 7
FSBR	fused salt (fast) breeder reactor
FTU	fuel treatment unit
GC	(IAEA) General Conference
GDC	general design criteria
GEMS	Gibbs energy minimization software
GIF	Generation IV International Forum
GPC	generalized polynomial chaos
GPT	generalized perturbation theory
HALEU	high assay low enriched uranium
HFR	high flux reactor
HPD	hydrogen proportional detectors
HSR	hard spectrum reactor
HTR	high temperature reactor
HTS	high temperature molten salt
HX	heat exchanger
IABAT	impact of the accelerator based technologies
IAEA	International Atomic Energy Agency
ICP-AES	inductively coupled plasma- atomic emission spectrometry
ICP-MS	inductively coupled plasma- mass spectrometry
IFK	Institut für Festkörper-Kernphysik (Germany)
IFR	integral fast reactor
IGC	intergranular corrosion
IHX	intermediate heat exchanger

IMSBR	Indian molten salt breeder reactor
IMSR	integral MSR
IN2P3	National Institute of Nuclear Physics and Particle Physics (Institut national de physique nucléaire et de physique des particules) (France)
IRFU	Institute of research into the fundamental laws of the Universe (CEA, France)
IRSN	Radioprotection and Nuclear Safety Institute (Institut de radioprotection et de sûreté nucléaire) (France)
ISAM	integrated safety assessment methodology
ISOL	isotope separation on-line
ISTC	International Science and Technical Centre (Kazakhstan)
ITER	international thermonuclear experimental reactor
ITMSF	international thorium molten-salt forum
JAERI	Japan Atomic Energy Research Institute
JEFF	Joint Evaluated Fission and Fusion
JENDL	Japanese evaluated nuclear data library
JRC	Joint Research Centre (European Commission)
JRCMSD	Joint Research Centre Molten Salt Database
KEMA	Keuring van Elektrotechnische Materialen te Arnhem (Netherlands)
KEMS	Knudsen effusion mass spectrometry
KI	Kurchatov Institute (Russia)
KIT	Karlsruhe Institute of Technology
KP-FHR	Kairos Power fluoride-salt-cooled high temperature reactor
KSTR	KEMA suspension test reactor
LAF	laser flash
LANL	Los Alamos National Laboratory
LBE	linear Boltzmann equation
LDA	laser doppler anemometry
LEU	low enriched uranium
LFTR	liquid-fluoride thorium reactor
LGC	Chemical Engineering Laboratory (Toulouse, France)
LMFR	liquid metal fast reactor
Ln(s)	lanthanide(s)
LOD	lines of defence
LPSC	Laboratoire de Physique Subatomique et de Cosmologie (Subatomic Physics and Cosmology Laboratory) (France)
LWR	light water reactor
MA	minor actinides

MARS	minor actinides recycling in salts (project 2010-2013)
MATLAB	matrix laboratory
MCC	Mining and Chemical Combine (Russia)
MCFBR	molten chloride fast breeder reactor
MCFR	molten chloride fast reactor
MCSFR	molten chloride salt fast reactor
MCNP	Monte Carlo n-type particle
MCPL	Monte Carlo particle lists
MD	molecular dynamics
MEAM	modified embedded atom model
MIT	Massachusetts Institute of Technology
MLD	master logic diagram
MONICR	MOlybdenum-NiCkel-CRoium (alloy)
MOSART	molten salt actinide recycler transforming system
MOSEL	molten salt epithermal
MOST	molten salt reactor technology (project)
MOU	memorandum of understanding
MOX	mixed oxide fuel
MP	multiphysics
MPM	multiphysics modelling
MSBE	molten salt breeder experiment
MSBR	molten salt breeder reactor
MSDR	molten salt demonstration reactor
MSFBR	molten salt fast breeder reactor
MSFR	molten salt fast reactor (such as the one developed in CNRS, France)
MSR	molten salt reactor
MSRE	molten salt reactor experiment
MSRP	molten salt reactor program
MSTW	molten salt thermal wasteburner
MW	megawatt
NB	New Brunswick, Canada
NE	Office of Nuclear Energy (US Department of Energy)
NEA	Nuclear Energy Agency (OECD)
NEEDS	nucléaire, énergie, environnement, déchets, société (France)
NEI	Nuclear Energy Institute (US)
NEMO	Nuclear Engineering MOdelling (group) at Politecnico di Torino, Italy
NEPA	nuclear energy for the propulsion of aircraft (project)
NES	(PSI) Nuclear Energy and Safety Division
NNCL	nitrate natural circulation experiment
NNSA	(China) National Nuclear Security Agency

NPP	nuclear power plant
NPTDS	(IAEA's) Nuclear Power Technology Development Section
NRC	(US) Nuclear Regulatory Commission
NRG	Nuclear Research and Consultancy Group, Netherlands
OECD	Organization for Economic Co-operation and Development
OF	OpenFOAM
OPG	Ontario Power Generation
ORNL	Oak Ridge National Laboratory
ORSORT	Oak Ridge School Of Reactor Technology
OVC	(salt) over-cooling
QA	quality assurance
PB-FHR	pebble bed fluoride-salt-cooled high-temperature reactor
PBR	pebble bed reactor
PIE	postulated initiating event
PIRT	phenomenon identification and ranking table
PIV	particle image velocimetry
POD	proper orthogonal decomposition
POLIMI	Politecnico di Milano, Italy
POLITO	Politecnico di Torino, Italy
P&T	partitioning and transmutation (of nuclear waste)
pcm	percent mille (unit of reactivity)
ppm	parts per million
PR&PP	proliferation resistance and physical protection
PSI	Paul Scherrer Institut (Switzerland)
pSSC	(GIF) provisional System Steering Committee
PV	photovoltaic
PWR	pressurized water reactor
R&D	research and development
RANS	Reynolds-averaged Navier Stokes
RE	reference electrode
REE	rare earth element
RELAP	reactor excursion and leak analysis program
RMG	reactor moderated by graphite
RMS	root mean square
ROM	reduced order modelling
RSWG	(GIF's) Risk and Safety Working Group
RW	radioactive waste

SAGA	salt gamma
SALIENT	salt irradiation experiment
SAMOFAR	safety assessment of the molten salt fast reactor (MSFR)
SAMOSAFER	severe accident modelling and safety assessment for fluid-fuel energy reactors
SIF	strategic innovation fund (Canada)
SINAP	Shanghai Institute of Applied Physics
SMR	small modular reactor
SOFT	salt reactor on site reprocessing fast converter task
SQ	significant quantity
SNF	spent nuclear fuel
SNSF	Swiss National Science Foundation
SSCs	structures, systems, and components
SSR	stable salt reactor
SWATH	salt at wall: thermal exchanges
TAS	total absorption spectroscopy
TCHR	thermal-convection heat removal
TDB	thermodynamic data base
TEM	transmission electron microscopy
TFM	transient fission matrix
TGADSC	thermo gravimetric analysis-differential scanning calorimetry
THORIMS-NES	thorium molten salt nuclear energy synergetic system
TLOP	total loss of power
TMC	total Monte Carlo (method)
TMSR	thorium molten salt reactor
TMSR-LF	liquid fuelled TMSR
TMSR-SF	solid fuelled TMSR
TRISO	tristructural isotropic
TRC	temperature reactivity coefficient
TRL	technical readiness level
TRS	(IAEA) Technical Report Series
TRU	transuranium (elements) (also known as transuranic (elements))
TU Delft	Delft University of Technology (Netherlands)
UCB	University of California, Berkeley
UK	United Kingdom
ULOFF	unprotected loss of fuel flow
ULOHS	unprotected loss of heat sink
UN	United Nations

UOX	uranium oxide fuel
UPOS	unprotected pump over-speed
UQ	uncertainty quantification
US	United States of America
V&V	verification and validation
VDR	(Canadian) vendor design review
VNIINM	A. A. Bochvar All-Russian Scientific Research Institute for Inorganic Materials
VTT	Valtion Teknillinen Tutkimuskeskus (State Technical Research Centre, Finland)
VVER	See WWER
WATSS	waste to stable salt (process)
WCR	water cooled reactor
WE	working electrode
WINS	World Institute for Nuclear Security
WWER	water-water energetic reactor
XRD	x-ray diffraction

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21-23 July, 2020

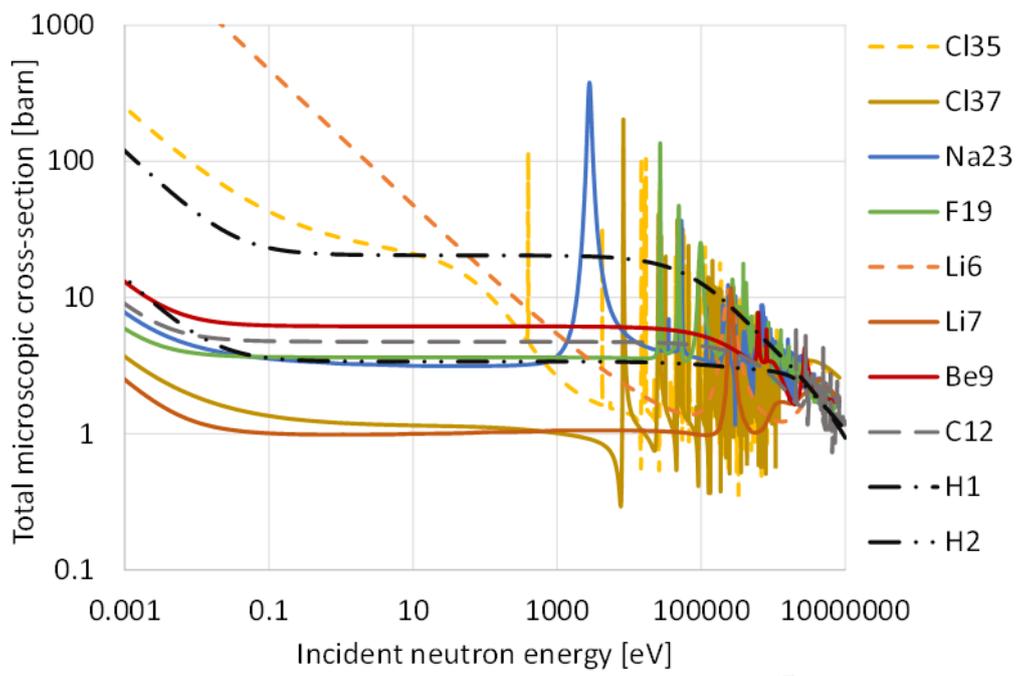
Technical Meeting

Vienna, Austria: 31 October – 4 November, 2016

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FIGURES

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FIG. 1. Total microscopic cross-section of selected nuclides.

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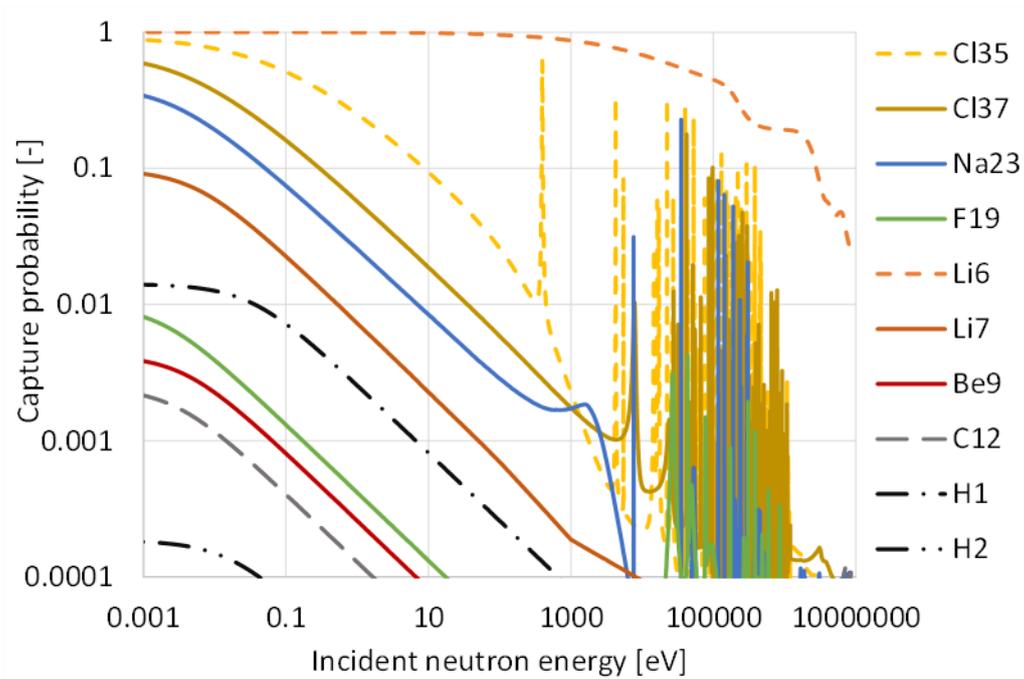


FIG. 2. Neutron capture probability of selected nuclides.

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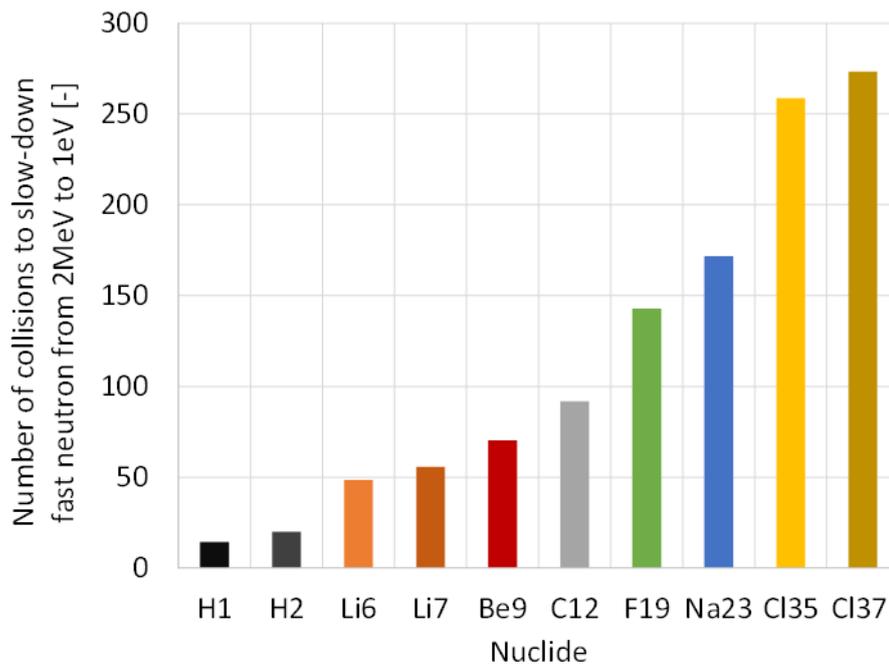
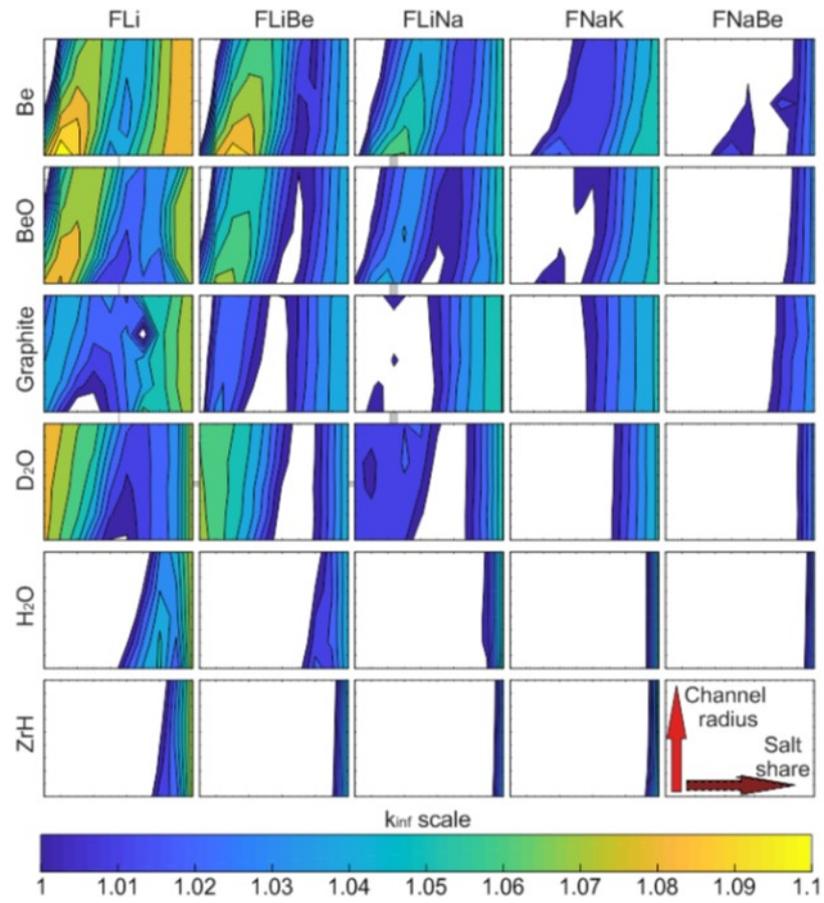


FIG. 3. Number of collisions to slow-down fast neutron from 2MeV to 1eV for selected nuclides.

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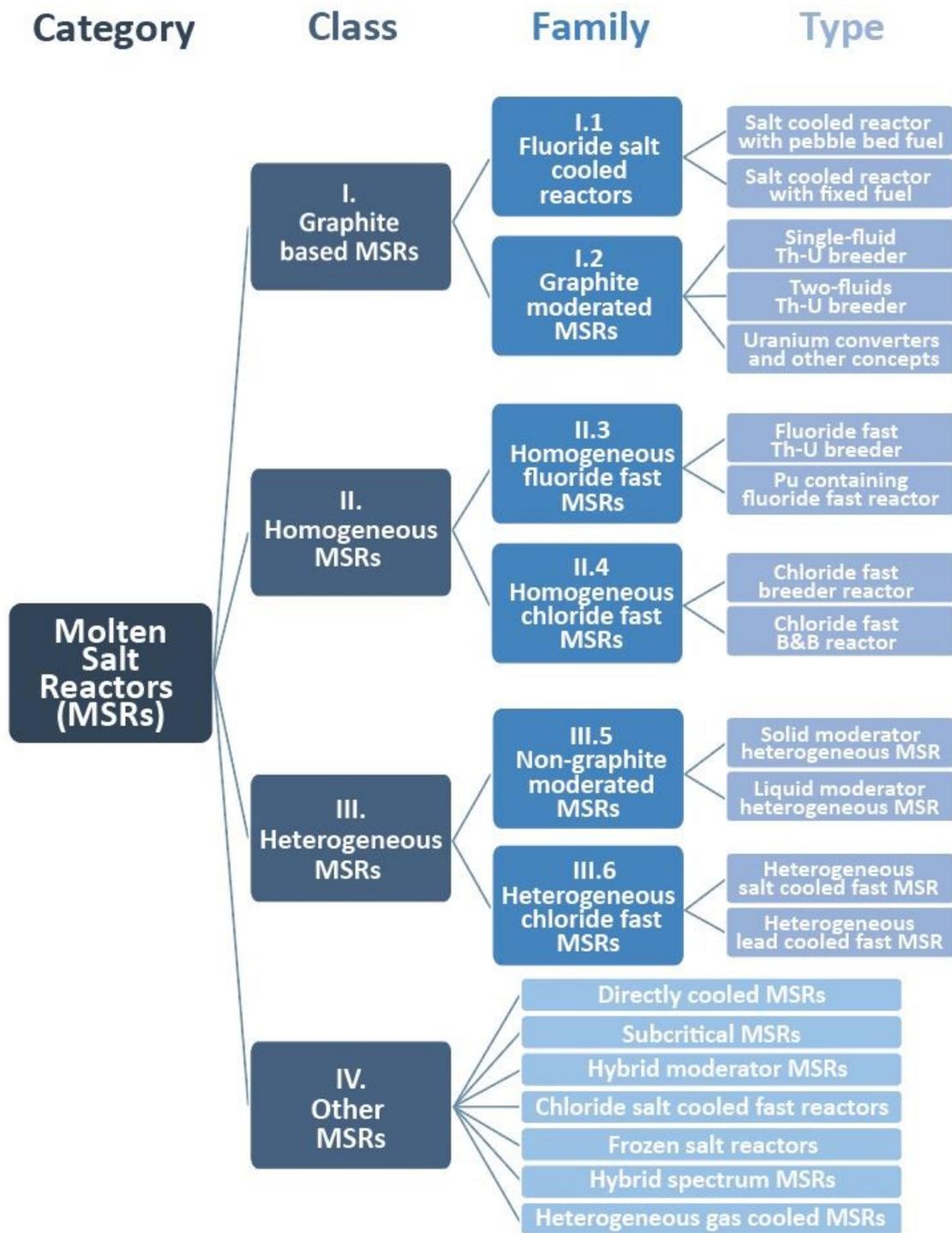
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FIG. 4. Equilibrium k_{inf} as a function of carrier salt type, moderator type, salt share in the core, and salt channel radius.

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PK

FIG. 5. First three layers of MSR taxonomy (tree view).

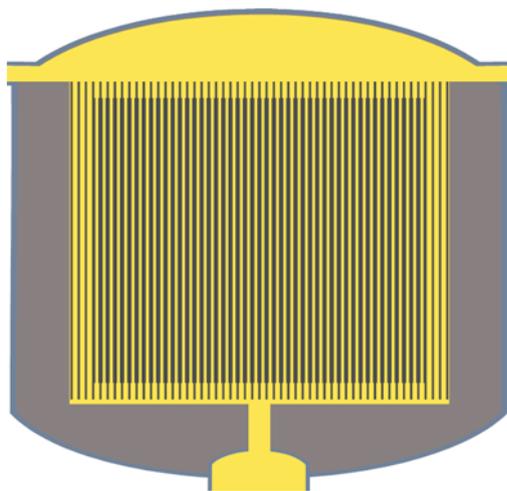


FIG. 6. Simplified figure of graphite moderated two-zone MSBR design proposed at ORNL in 1971.

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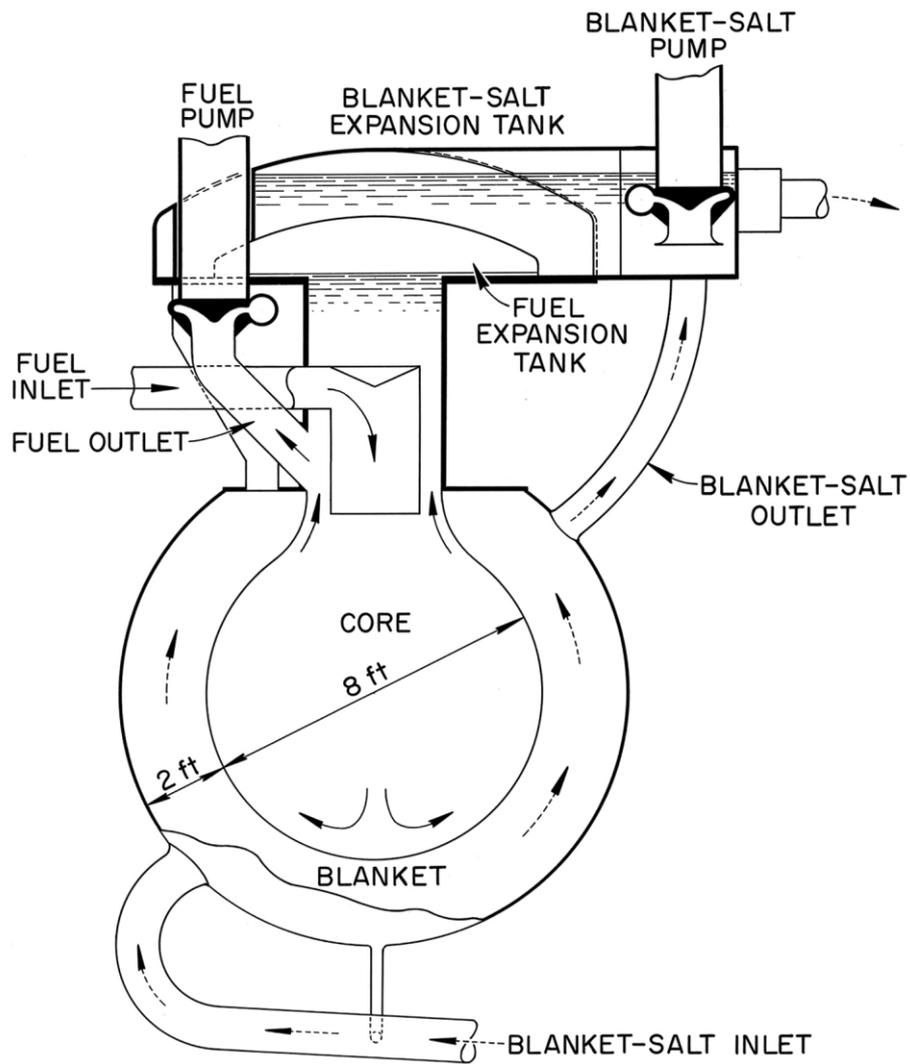
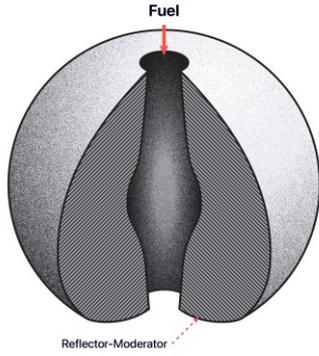
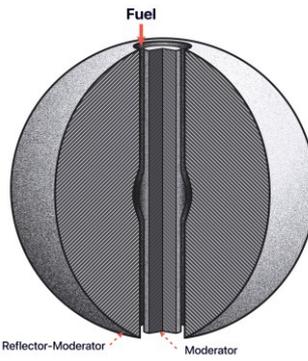


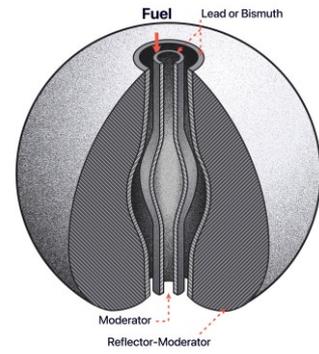
FIG. 7 Representative figure of Homogeneous fluoride fast MSRs concept proposed by ORNL in 1958 [113].



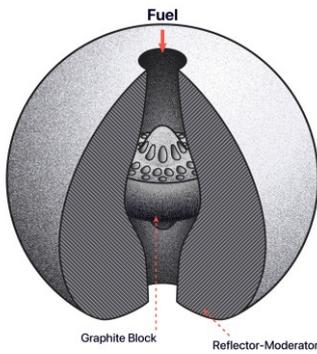
Simple two-region reactor core with thick, spherical shell of moderator surrounding a spherical chamber containing liquid fuel.



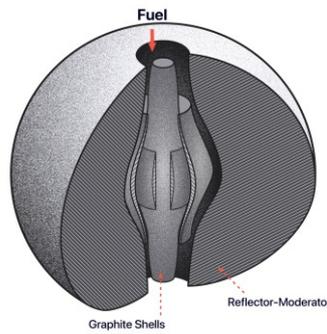
Three-region reactor core with central island of moderating materials.



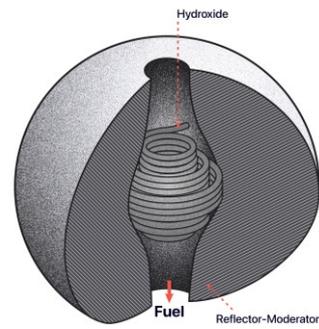
Five-region reactor core with provision for cooling reflector-moderator regions.



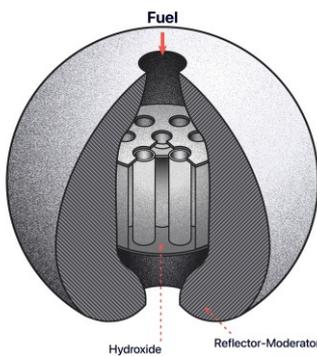
Reactor core with fuel channels in graphite block.



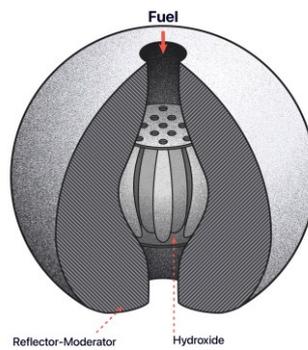
Reactor core with graphite shells in fuel channel.



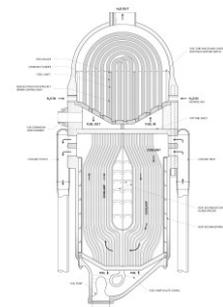
Fuel-moderator reactor core with coil tubes for circulating the moderator.



Fluid-moderated reactor core with straight-tube fuel passages and provisions for circulating moderator around fuel tubes.



Fluid-moderated reactor core with spheroidized fuel passages and provision for circulating moderator around fuel passages.



Reactor design for use with liquid moderator (water or hydroxide) and circulating fluoride fuel.

FIG. 8. Core concepts from ARE (Figures 51-58 and 64 from Ref. [76]).



FIG. 9. Fluoride salts and production facilities.

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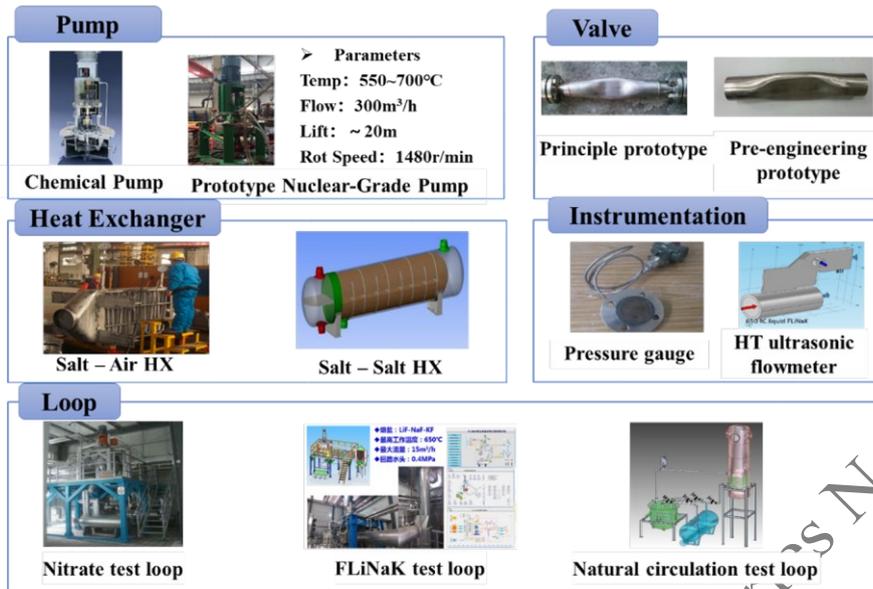


FIG. 10. Key equipment and molten salt loops developed in SINAP.

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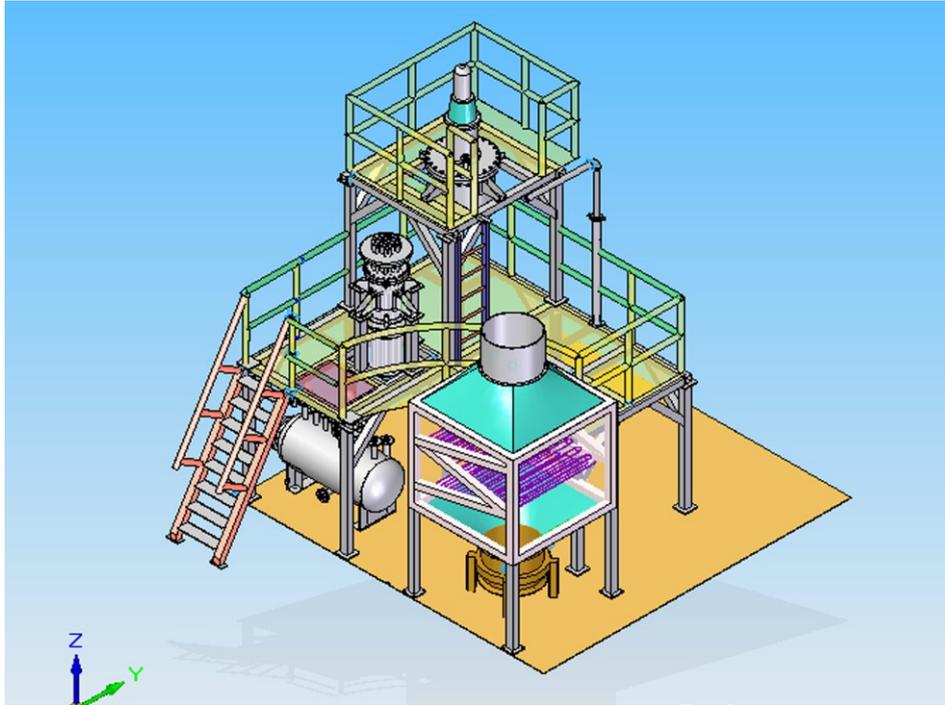


FIG. 11. Overall layout of HTS molten salt thermal experiment loop.

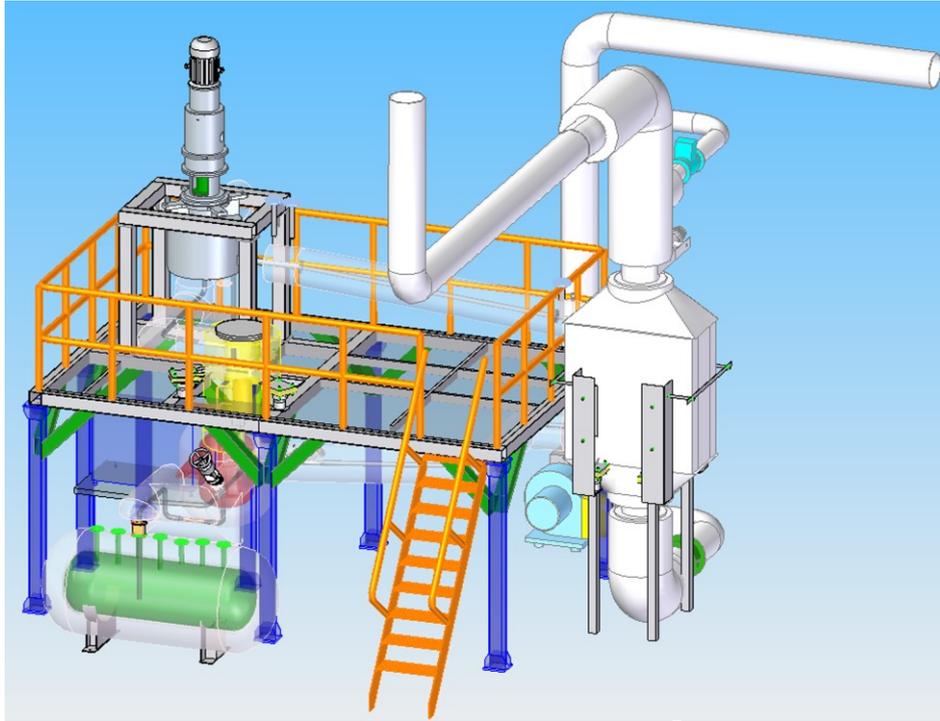


FIG. 12. Overall layout of the FLiNaK molten salt high temperature experiment loop.

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FIG. 13. Overall layout of NNCL loop.



FIG. 14. Radiochemistry facilities.

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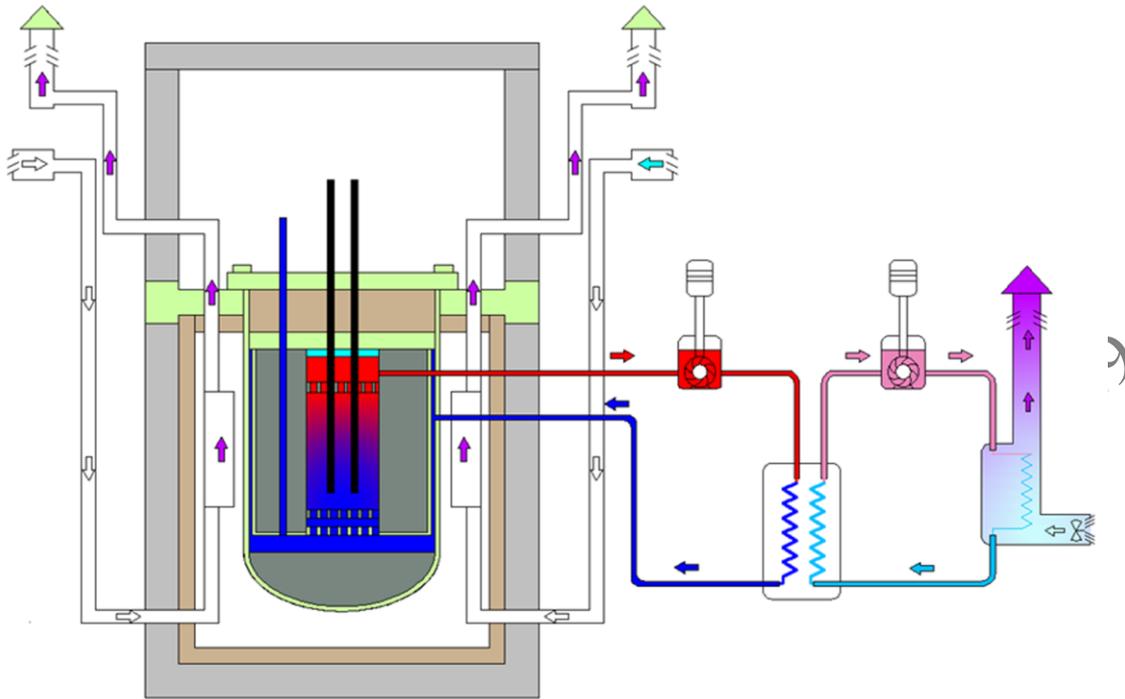


FIG. 15. Schematic layout of TMSR-0.

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FIG. 16. Layout of TMSR-0.

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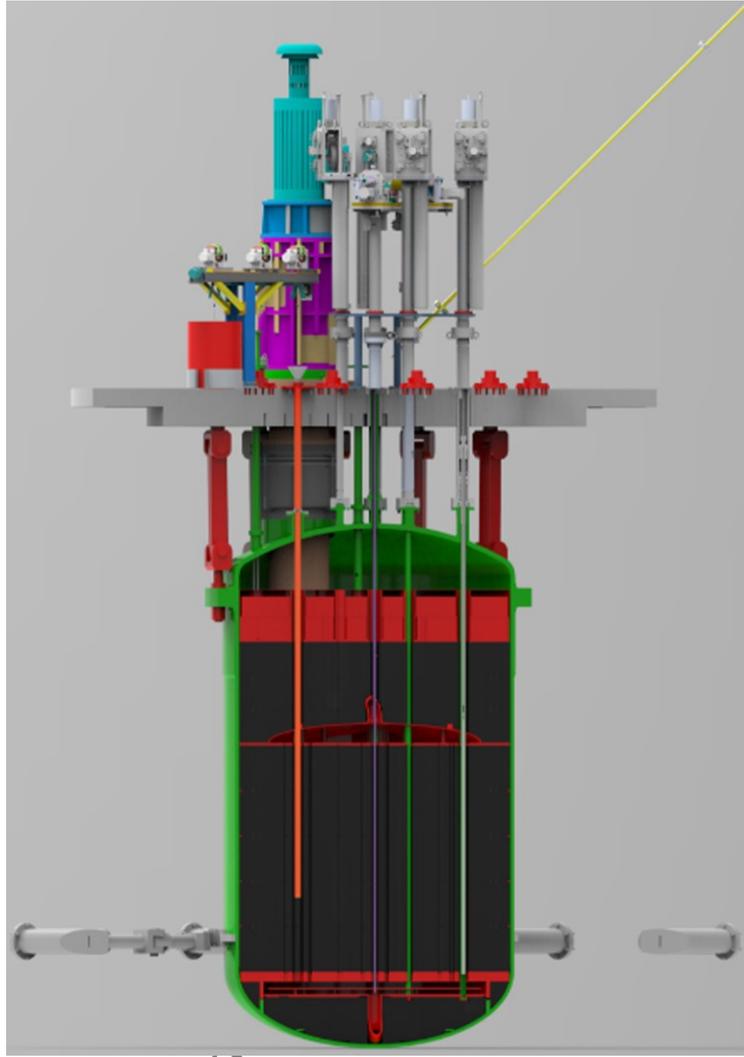


FIG. 17. Schematic layout of TMSR-LF1's reactor vessel and core.

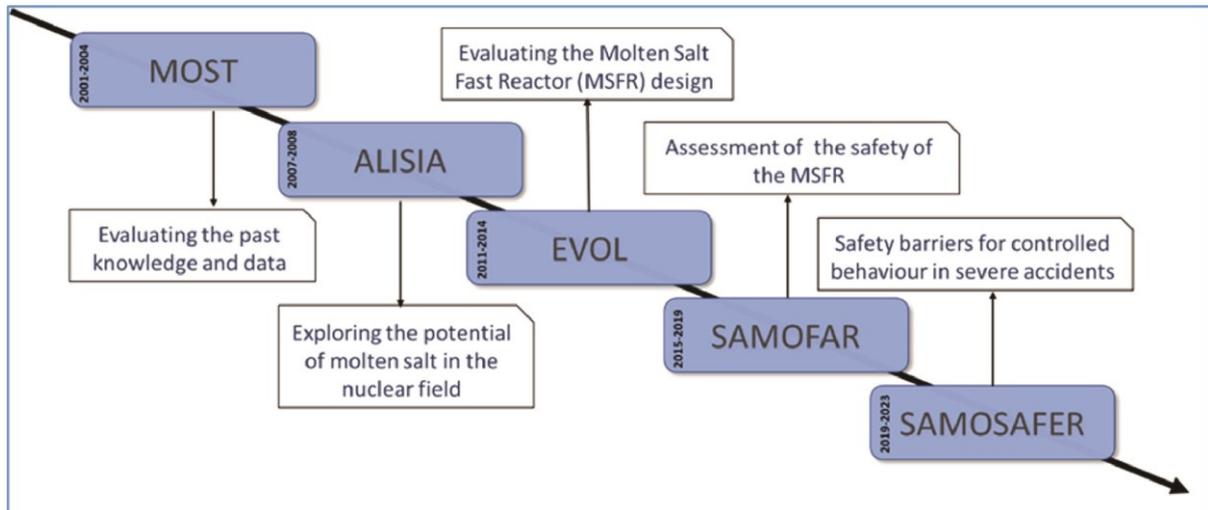


FIG. 18 Time evolution of EU-projects on MSR R&D.

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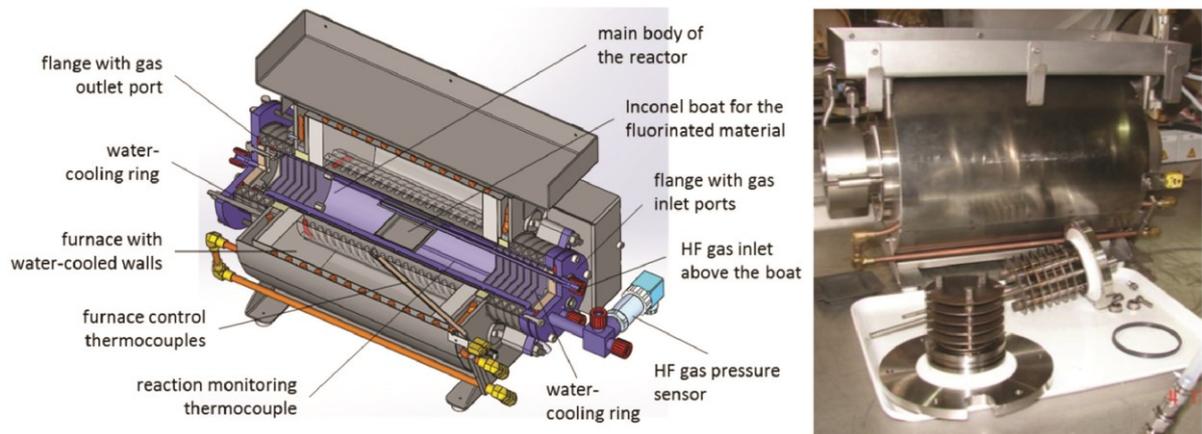


FIG. 19 A scheme and a photo of the fluorination reactor installed in JRC Karlsruhe.

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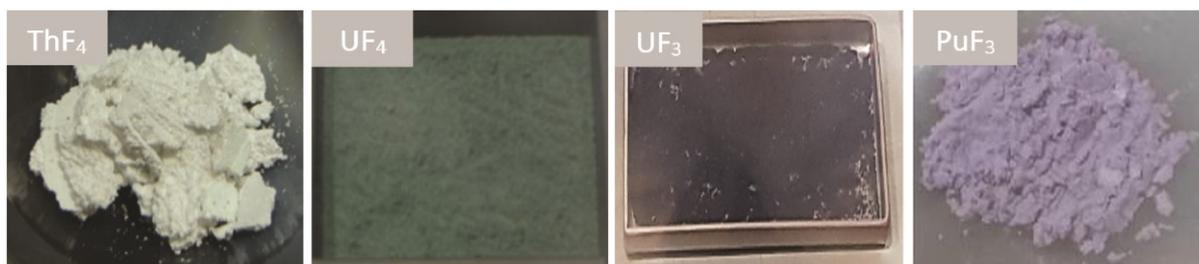


FIG. 20. Appearance of the pure actinides fluorides synthesised in JRC Karlsruhe

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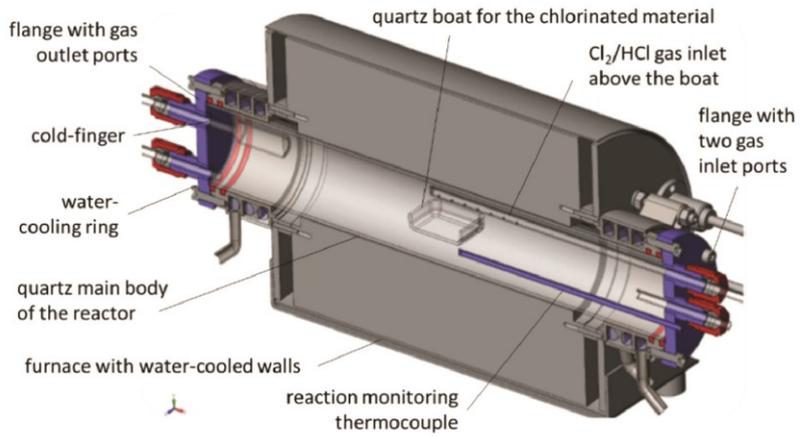


FIG. 21. A scheme and a photo of the chlorination reactor installed in IRC Karlsruhe.

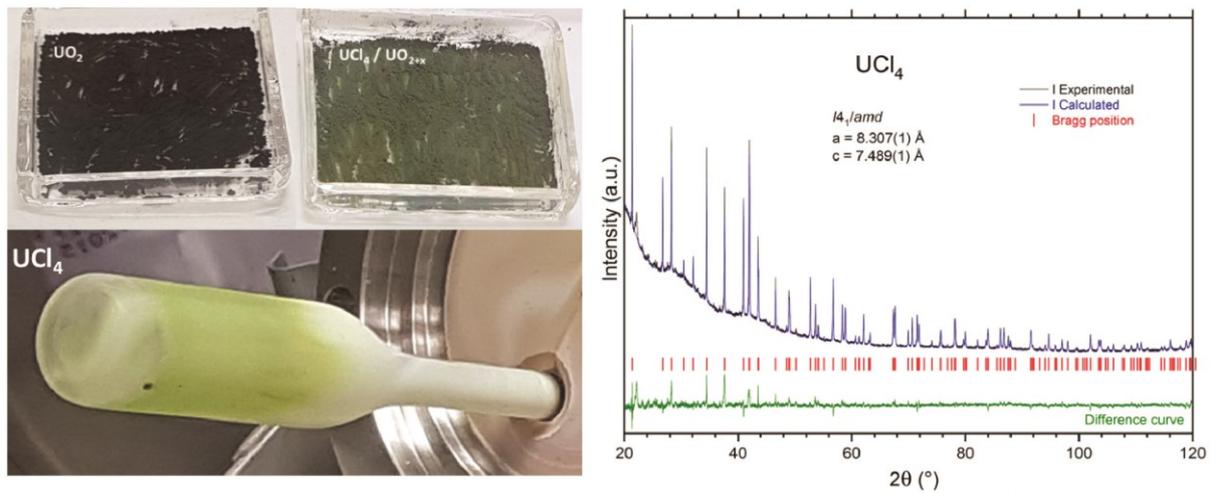


FIG. 22. Photos of the initial material, the chlorination and final products, and an XRD pattern of the phase-pure UCl_4 final product.

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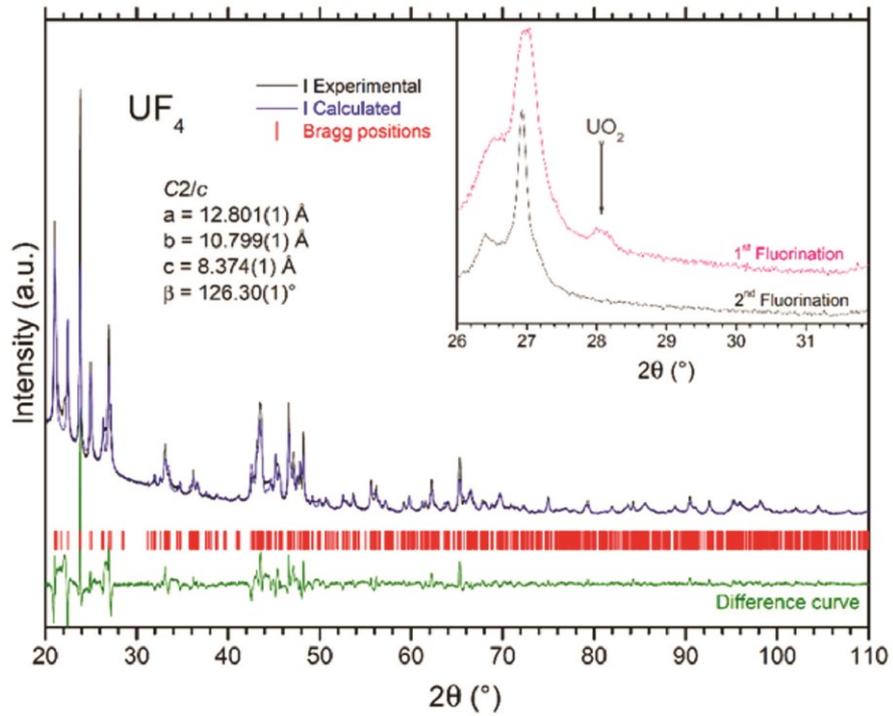


FIG. 23. XRD analysis of purified and not-purified UF_4 .

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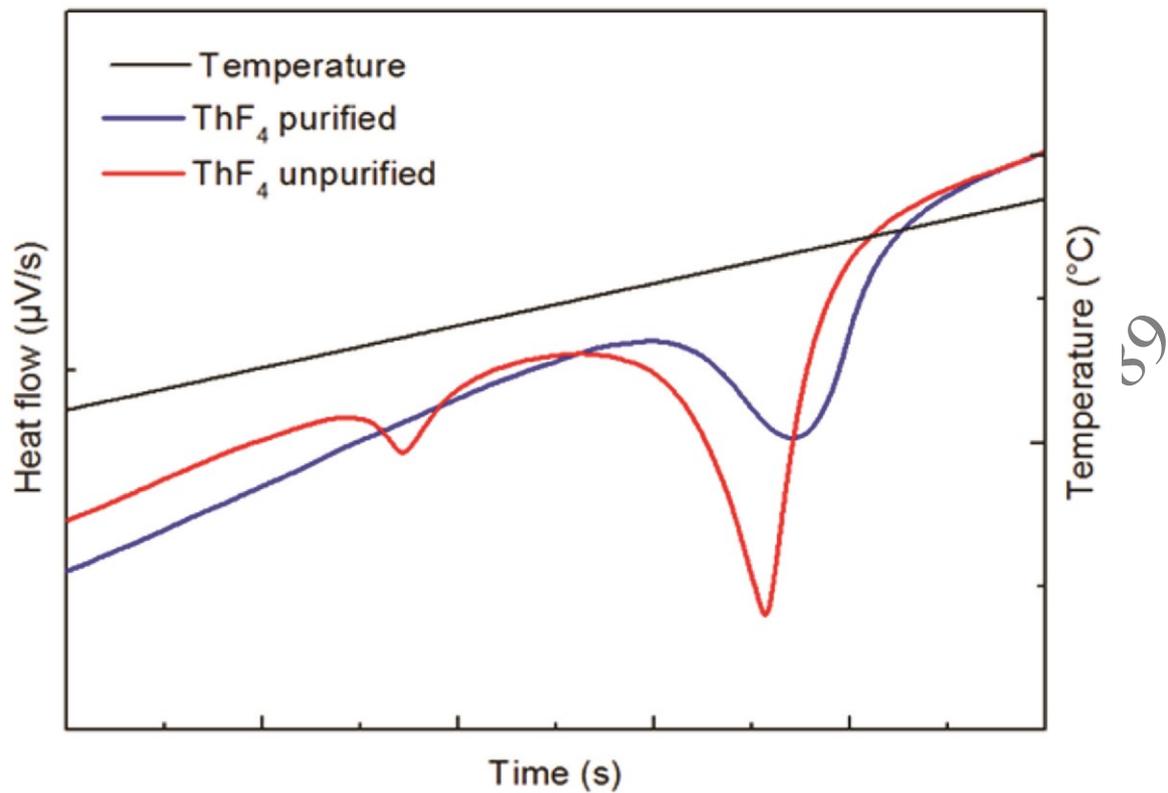


FIG. 24. Melting point determination of purified and not-purified ThF₄ determined by DSC.

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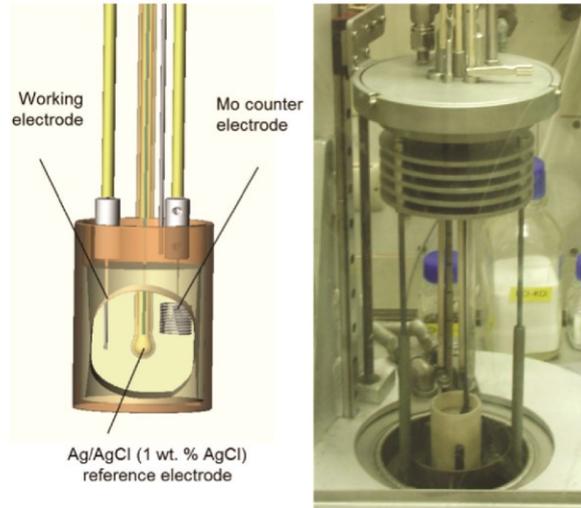
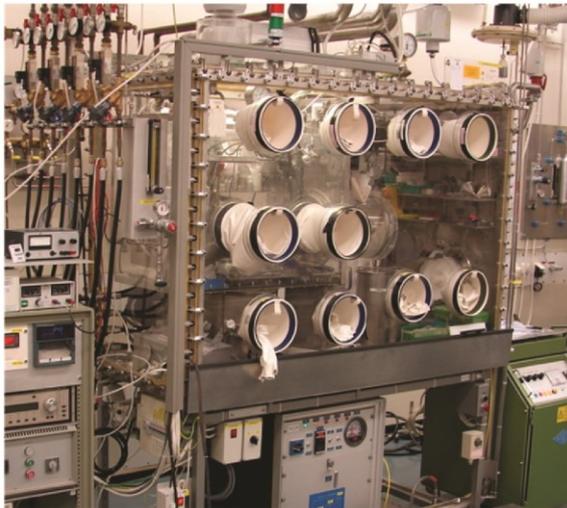


FIG. 25. Glove box for electrochemistry in molten chlorides installed in JRC Karlsruhe (left) and a scheme and a photo of the electrochemical cell (right).

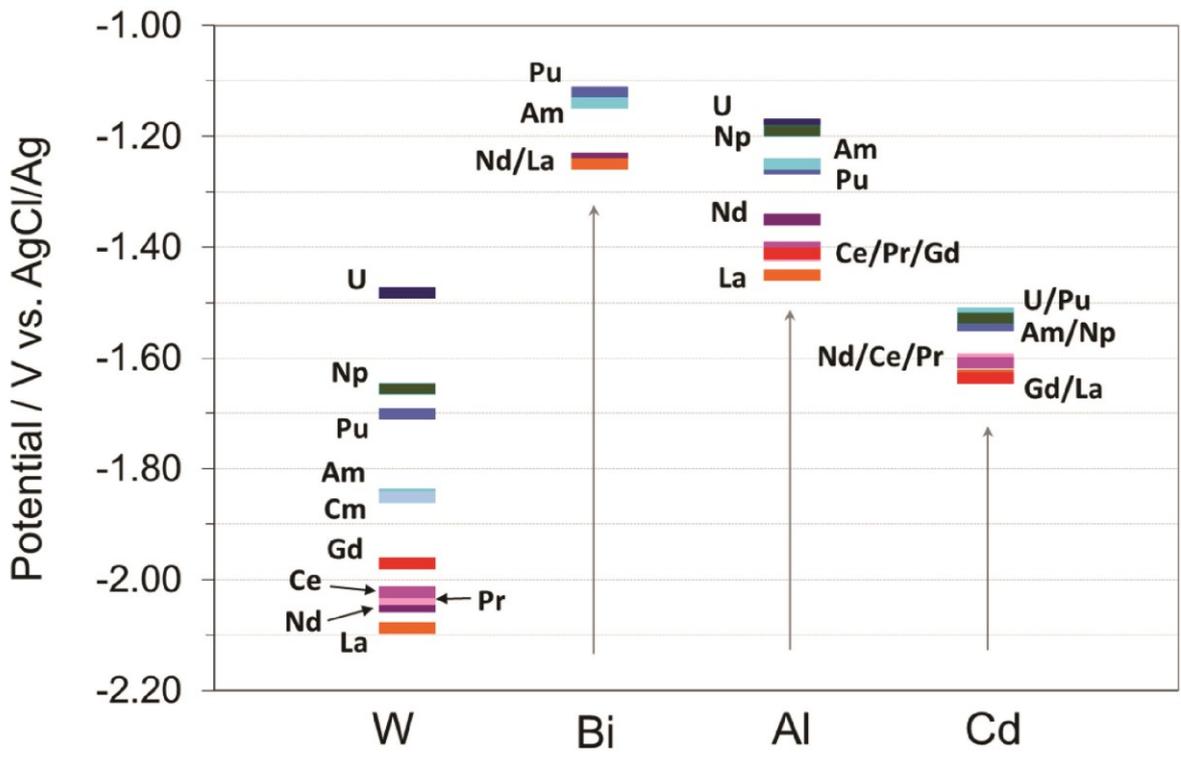


FIG. 26. Apparent electrochemical potentials measured in JRC Karlsruhe for various actinides and lanthanides using different working electrodes in LiCl-KCl eutectic at 450°C. The represented values are based on the measurements taken from Ref. [200].

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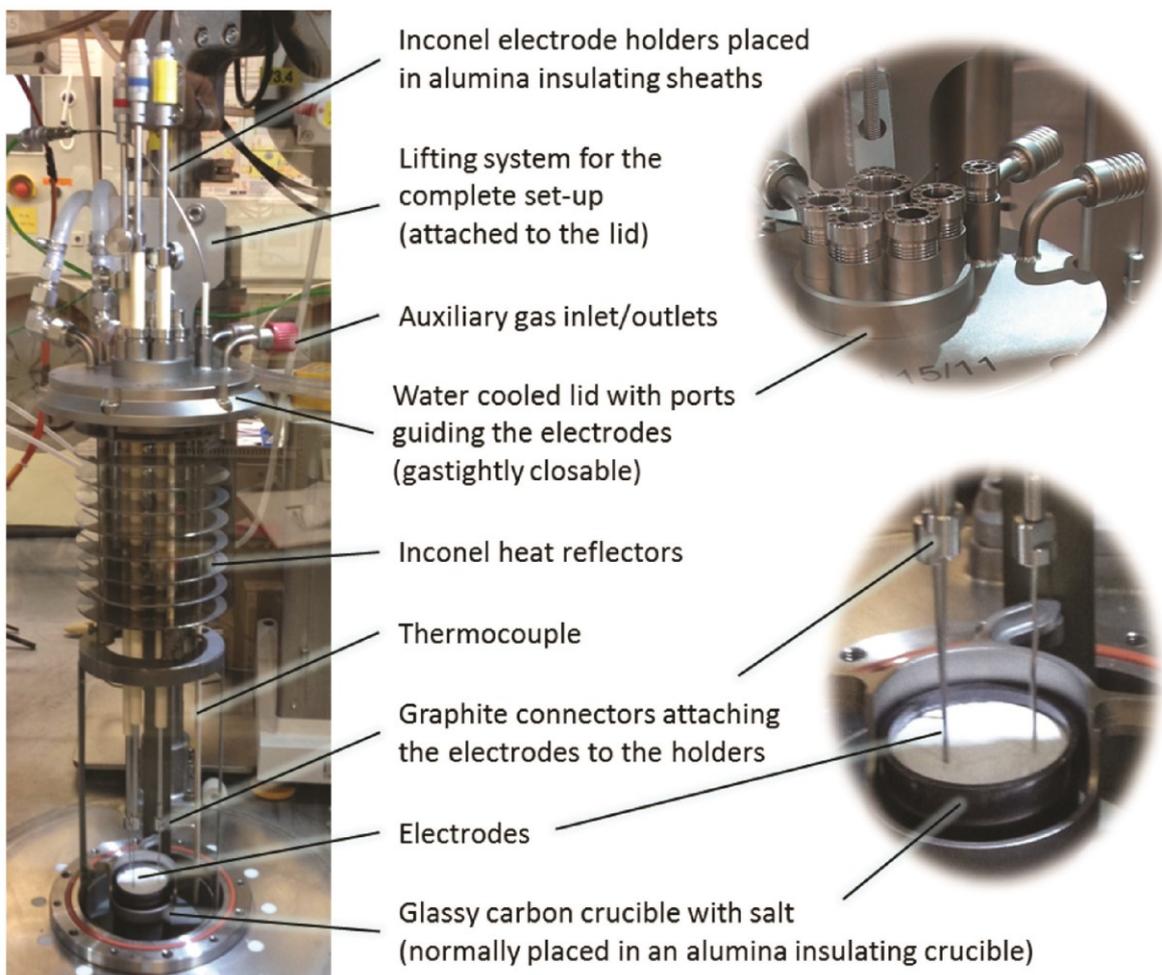


FIG. 27. A photo of the electrolyser installed in JRC Karlsruhe with description of the most important parts of the set-up and electrochemical cell.

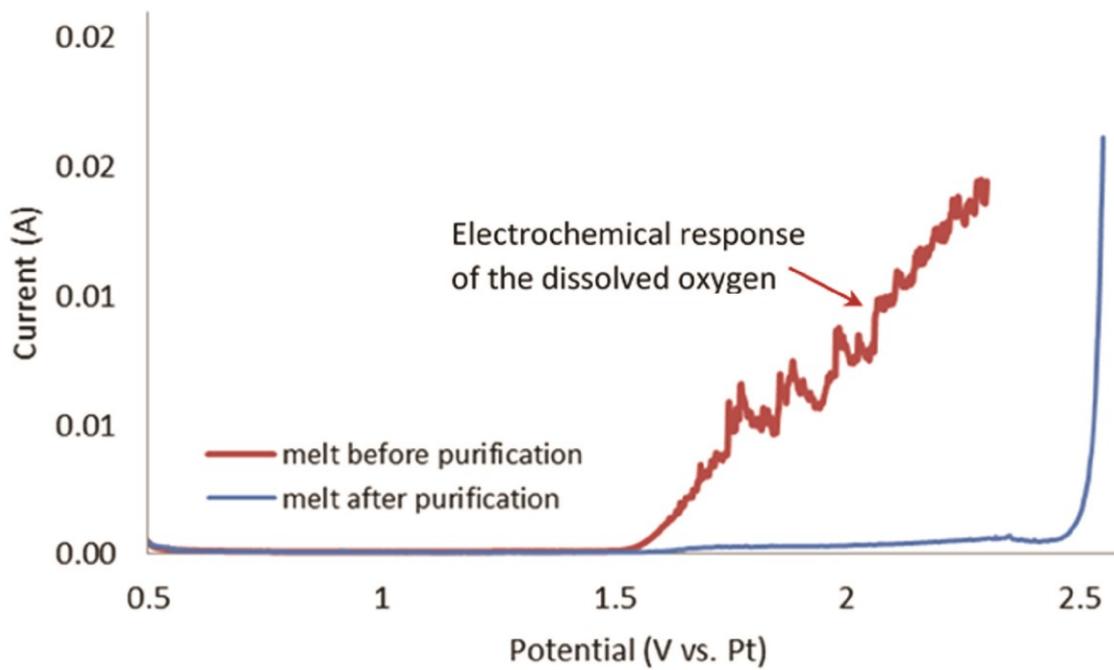


FIG. 28. Linear sweep voltammetry of LiF-CaF₂ eutectic melt before and after purification by HF gas. Working electrode: Au wire (cross section (S) = 0.08 cm²), counter electrode: glassy carbon rod, reference electrode: Pt wire, scan rate: 10 mV/S, temperature: 850°C.

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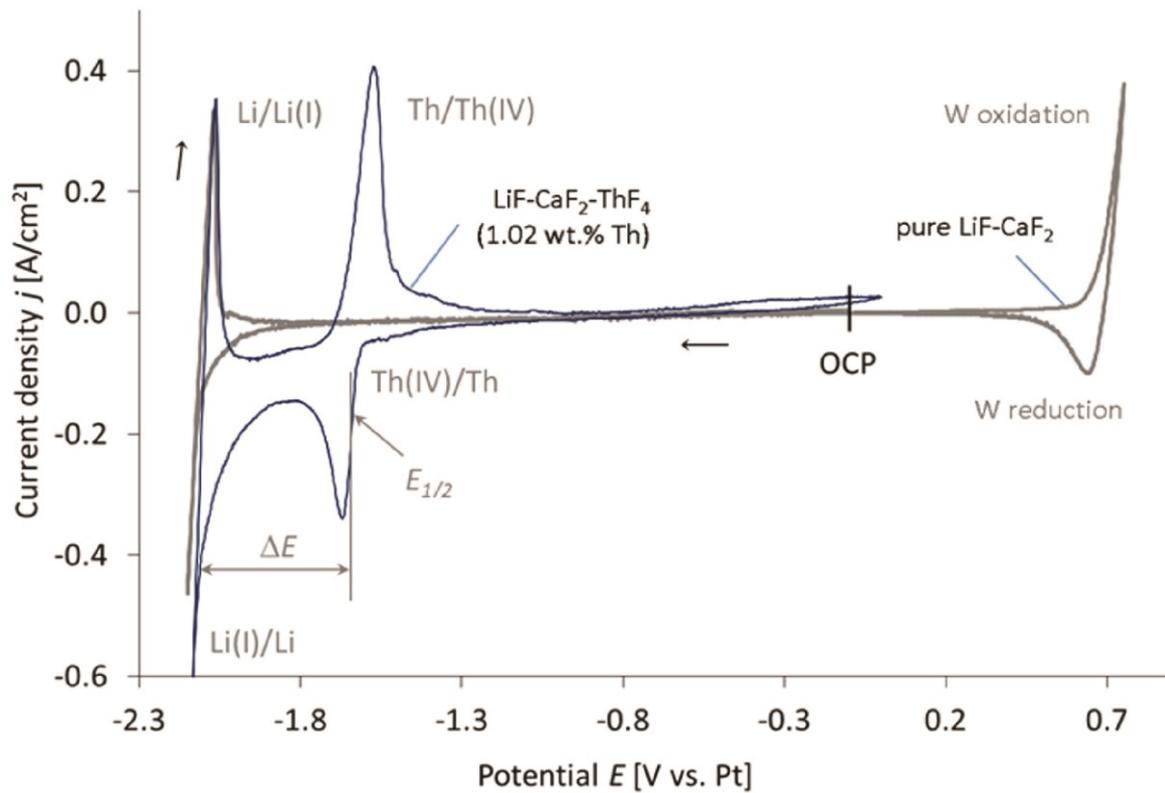


FIG. 29, Cyclic voltammetry of pure LiF-CaF₂ eutectic melt (thick line) and LiF-CaF₂-ThF₄ (0.18 mol.% ThF₄) melt showing graphical evaluation of the potential difference between ThF₄ and the least stable carrier melt constituent, LiF (working electrode: W, counter electrode: W, reference electrode: Pt, temperature: 850°C, scan rate: 100 mV/s).

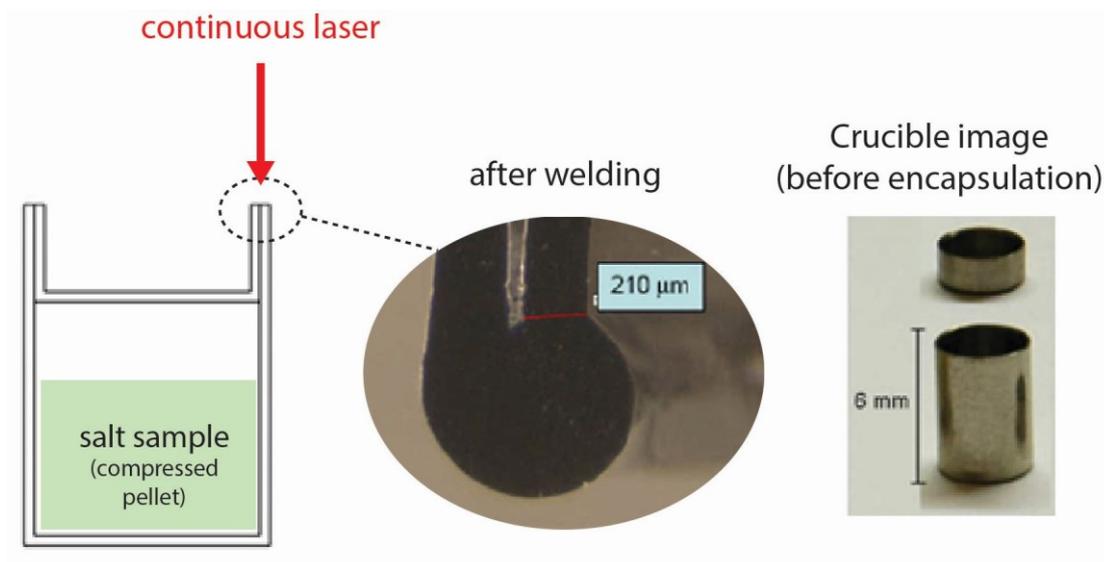


FIG. 30. A schematic presentation and a picture of the nickel crucible designed for the encapsulation of fluoride and chloride salt samples. The wall thickness is ~ 0.2 mm. Middle picture: Metallography of the welded edge of the crucible.

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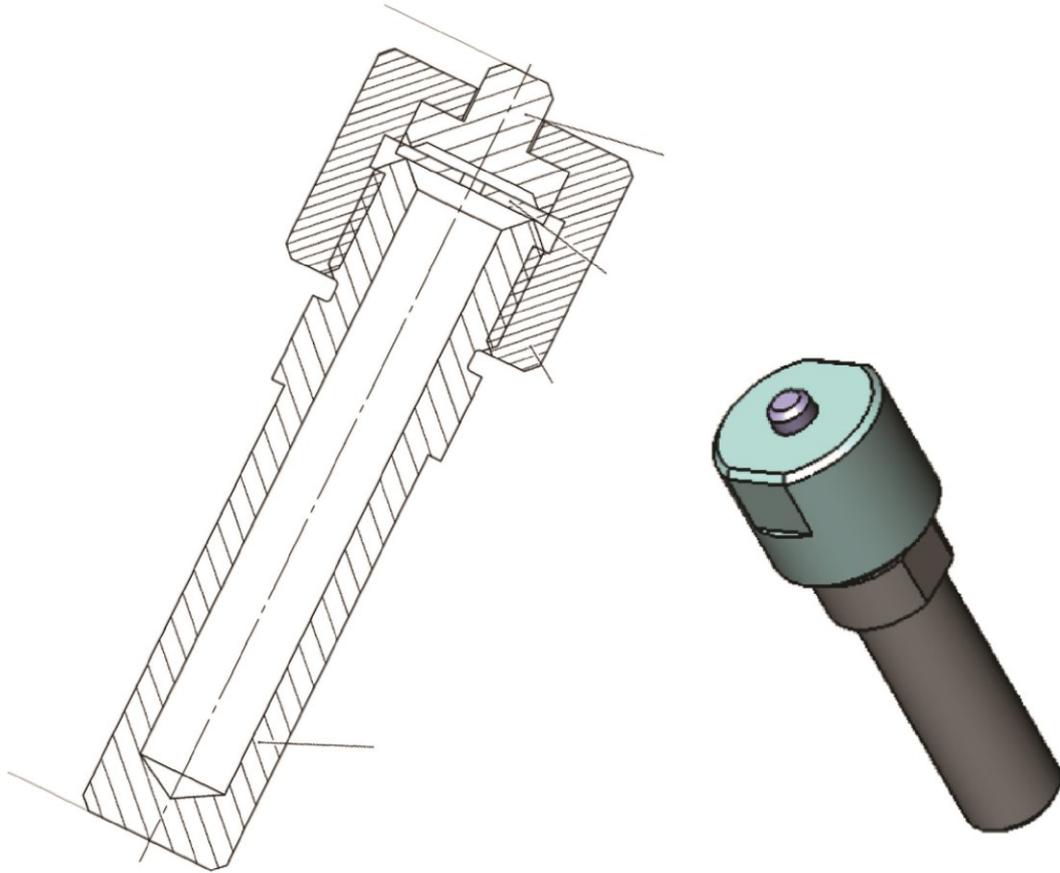


FIG. 31. Schematic representation of the DSC crucible developed at JRC for measurements of halide samples.

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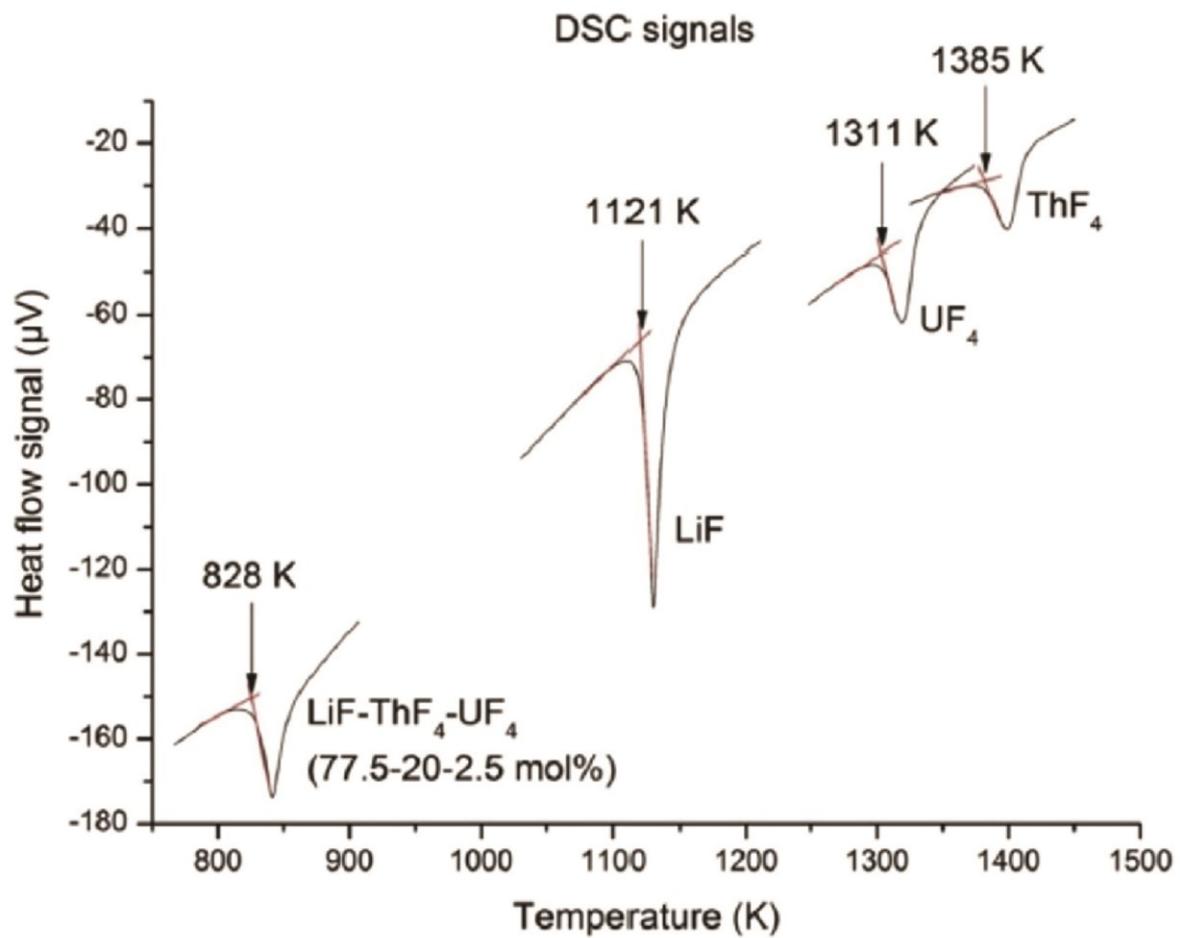


FIG. 32. A DSC output of the melting point determination of LiF , UF_4 and ThF_4 end-members and the $\text{LiF-ThF}_4\text{-UF}_4$ (77.5-20-2.5 mol%) MSFR fuel.

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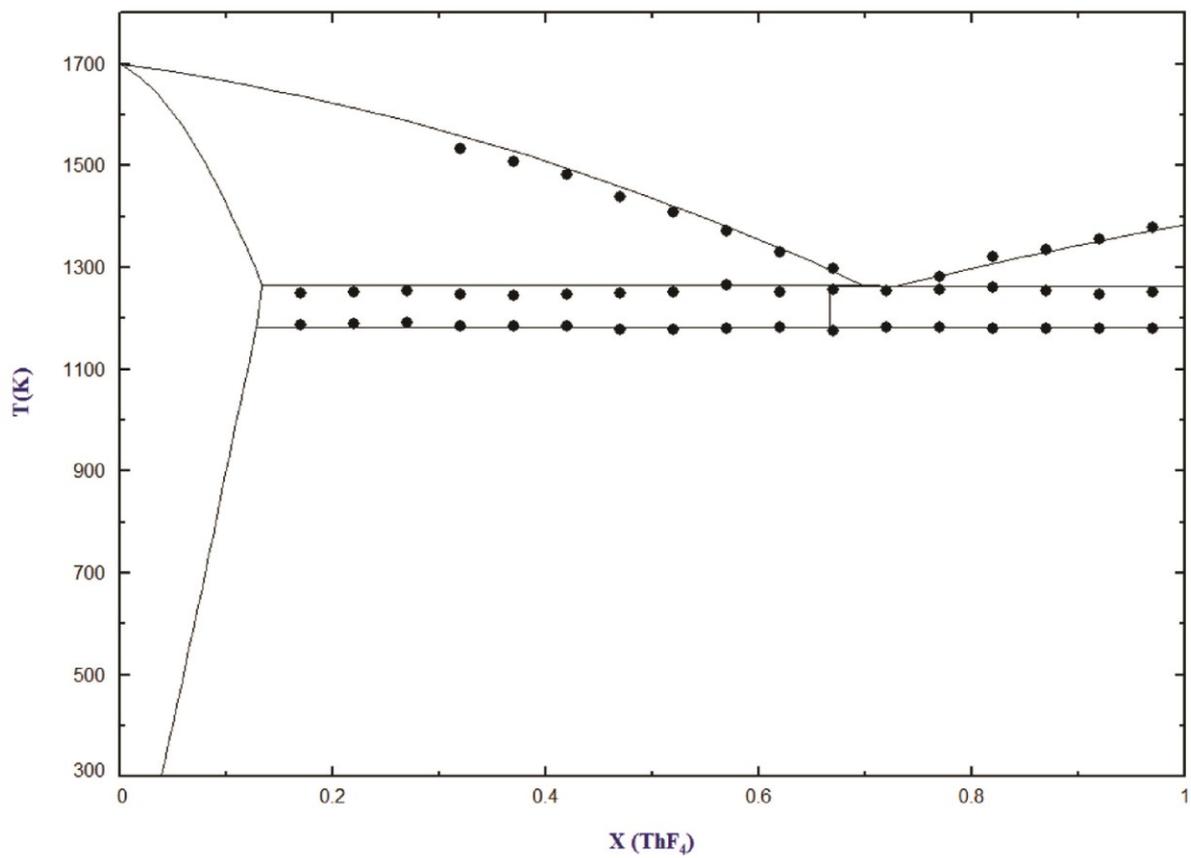


FIG. 33. The assessed PuF₃-ThF₄ phase diagram. The points are phase equilibrium points obtained by DSC.

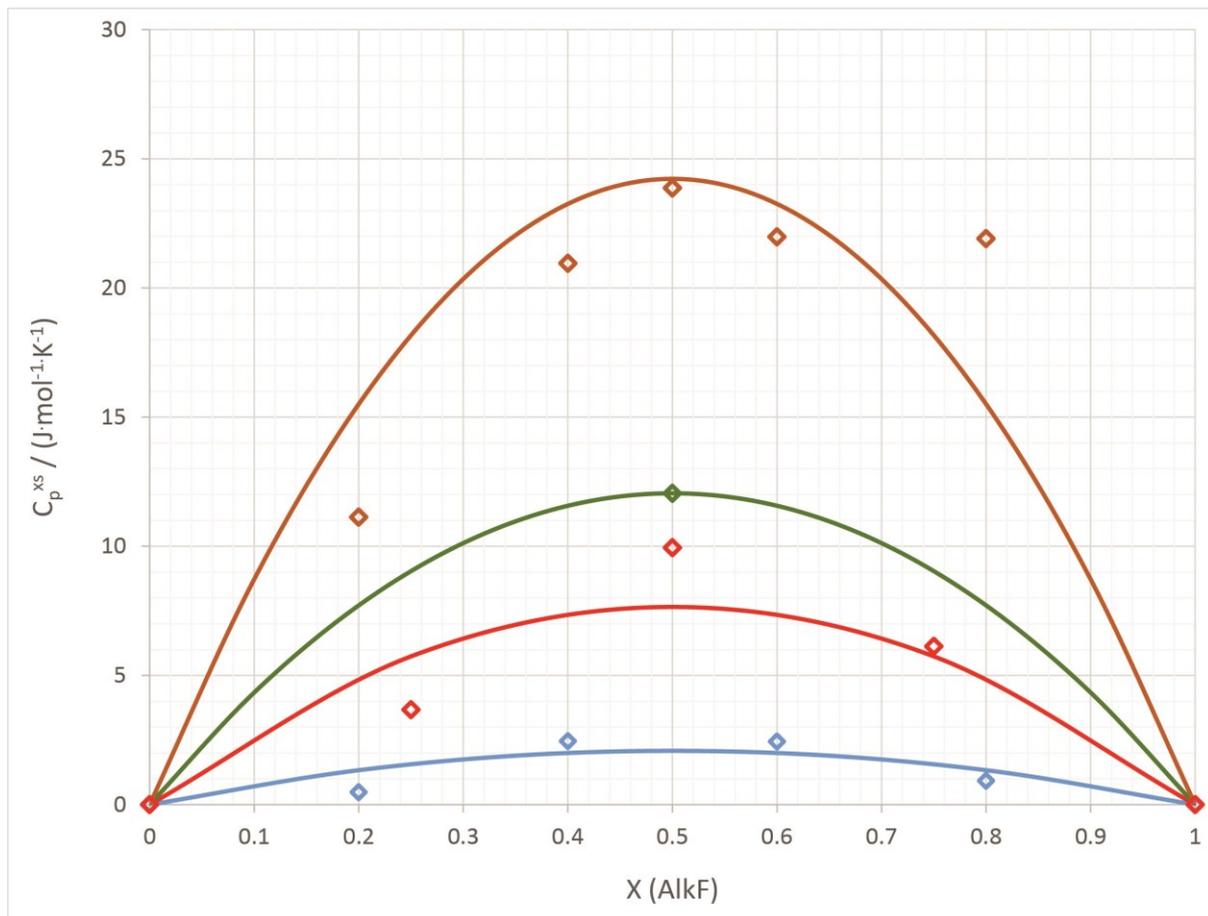


FIG. 34. Experimentally determined excess heat capacity as a function of molar ratio of mixed higher cation $X(\text{AlkF})$ for the $\text{LiF}-\text{AlkF}$ binary liquid solutions: $\text{LiF}-\text{NaF}$ (light blue); $\text{LiF}-\text{KF}$ (red); $\text{LiF}-\text{RbF}$ (green); $\text{LiF}-\text{CsF}$ (orange). The experimental data are taken from Ref. [212].

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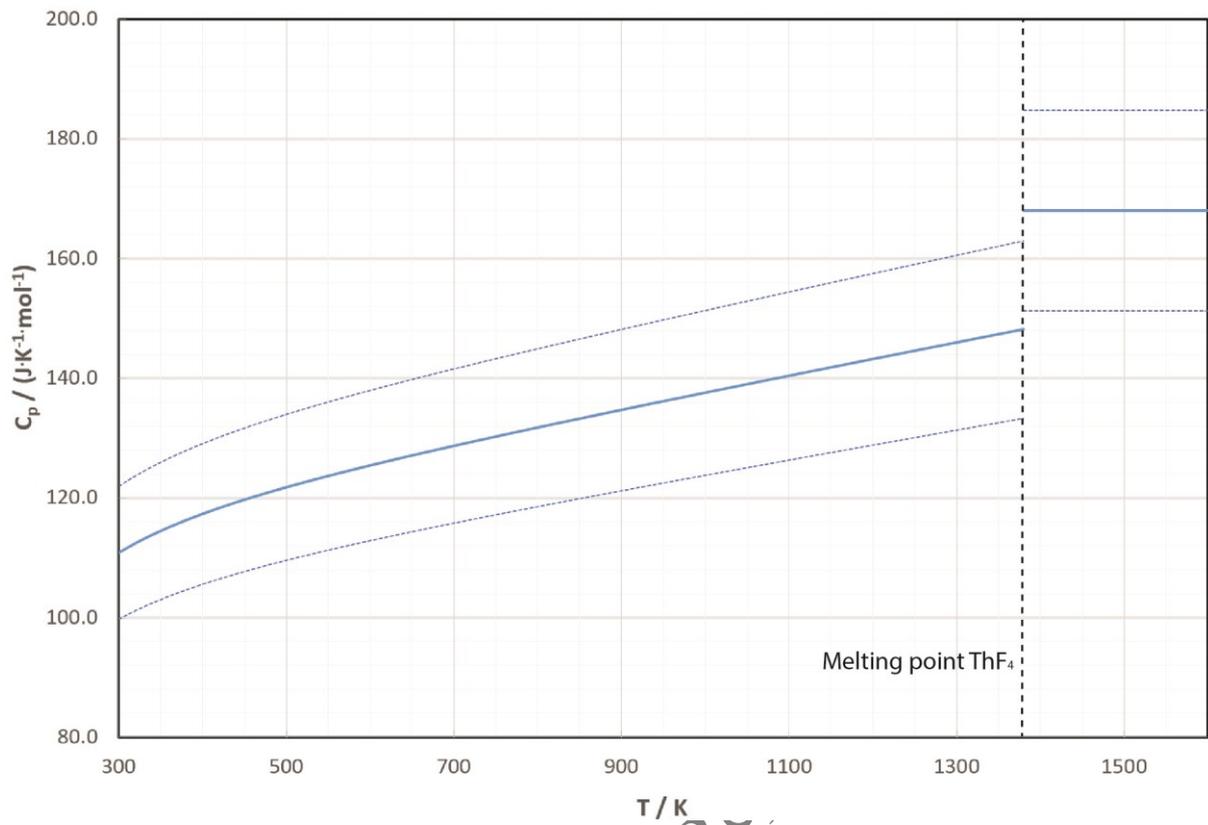


FIG. 35. Heat capacity of solid and liquid phases of ThF_4 obtained by DSC and Drop Calorimetry; data from Ref. [214].

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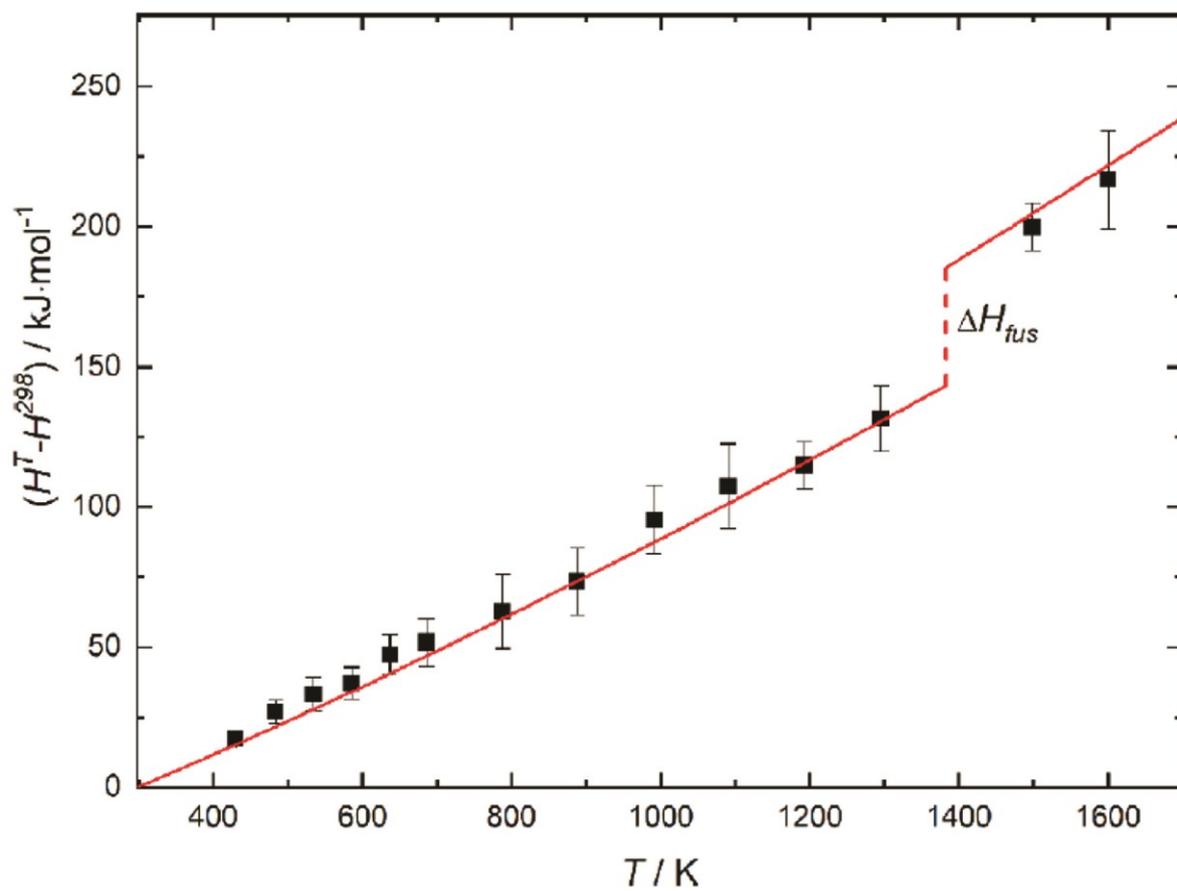


FIG. 36. Enthalpy increment data measured by Drop Calorimetry for ThF4 solid and liquid phases as a function of temperature (T) (in °K); data are from Ref. [214].

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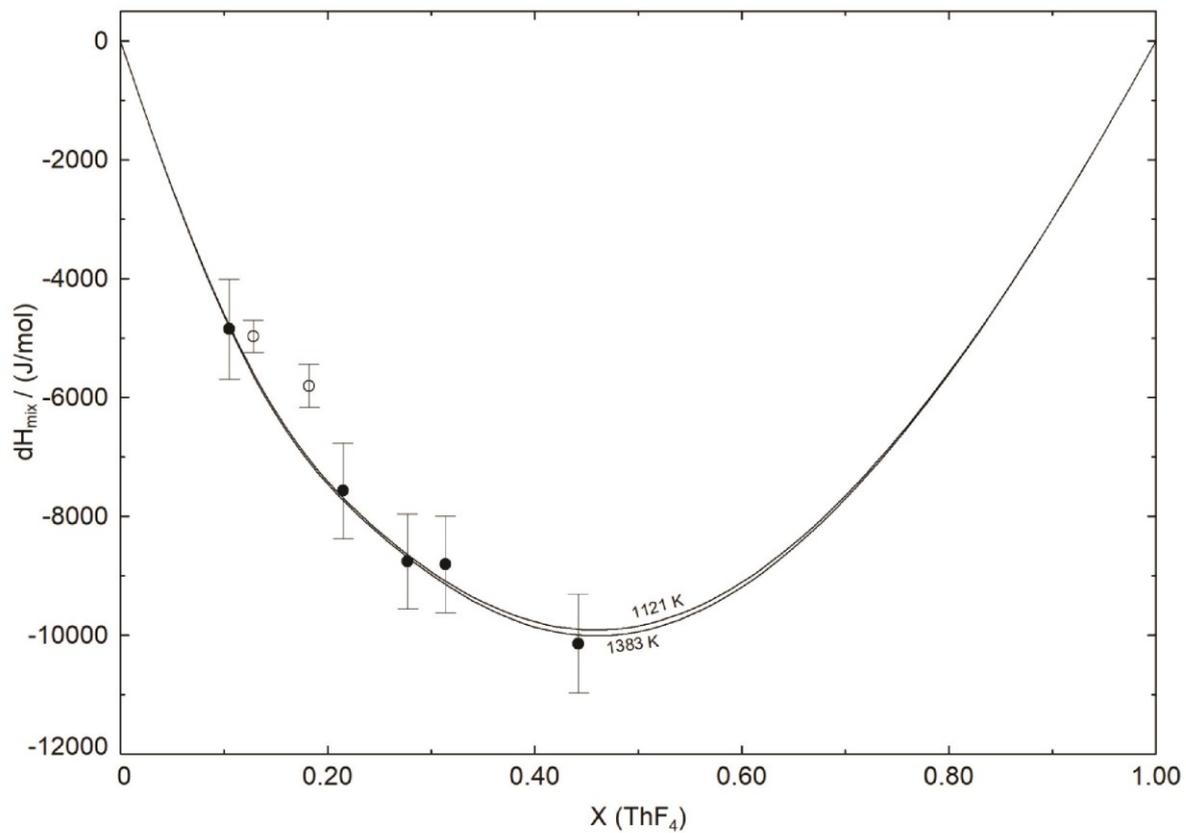


FIG. 37. Mixing enthalpy of the LiF-ThF4 binary system as determined by DSC; data are measurements taken from Ref. [209].

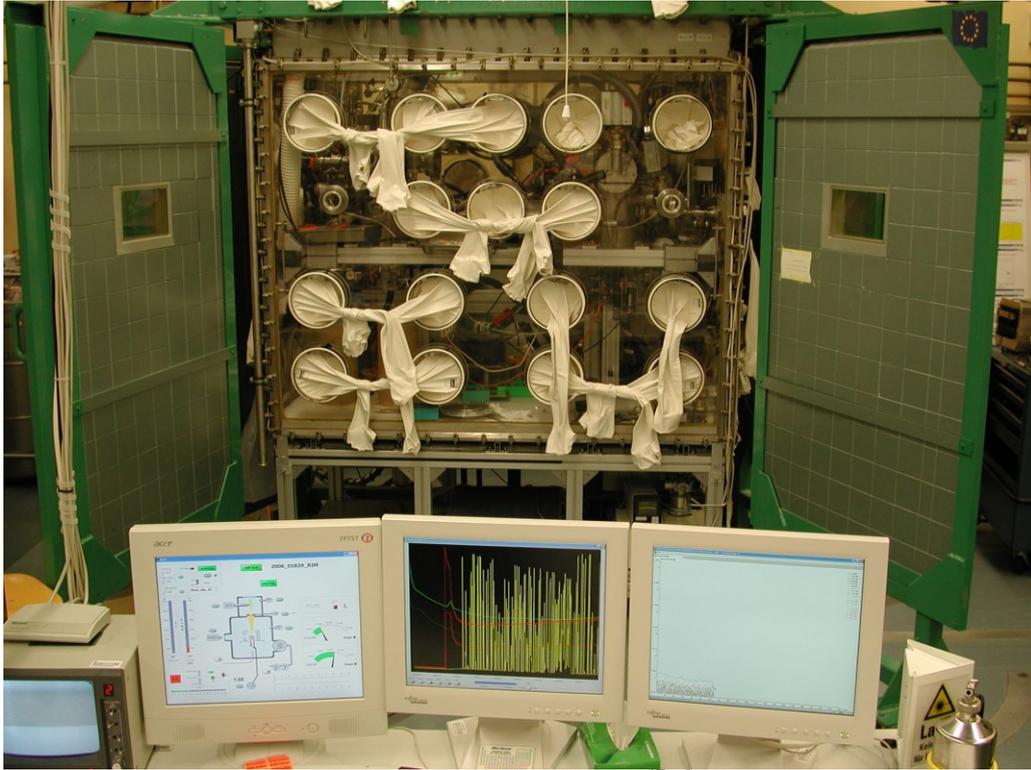


FIG. 38. Device for KEMS installed in alpha-tight glove box with external gamma shielding at JRC-Karlsruhe.

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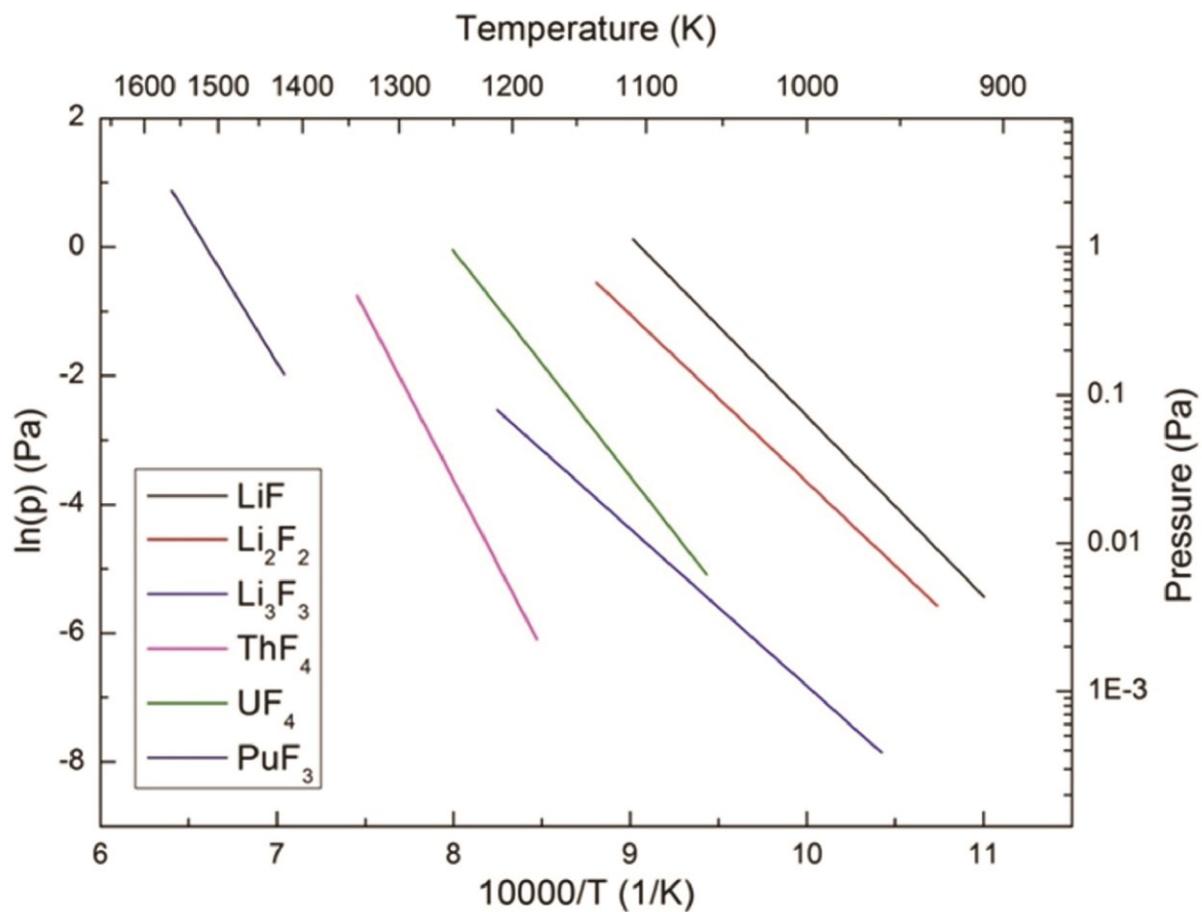


FIG. 39. Partial and total vapour pressures for the mixture LiF-ThF₄-UF₄-PuF₃ (77.5-6.6-12.3-3.6 mol%). Relative uncertainty on vapour pressure measurements is 50%.

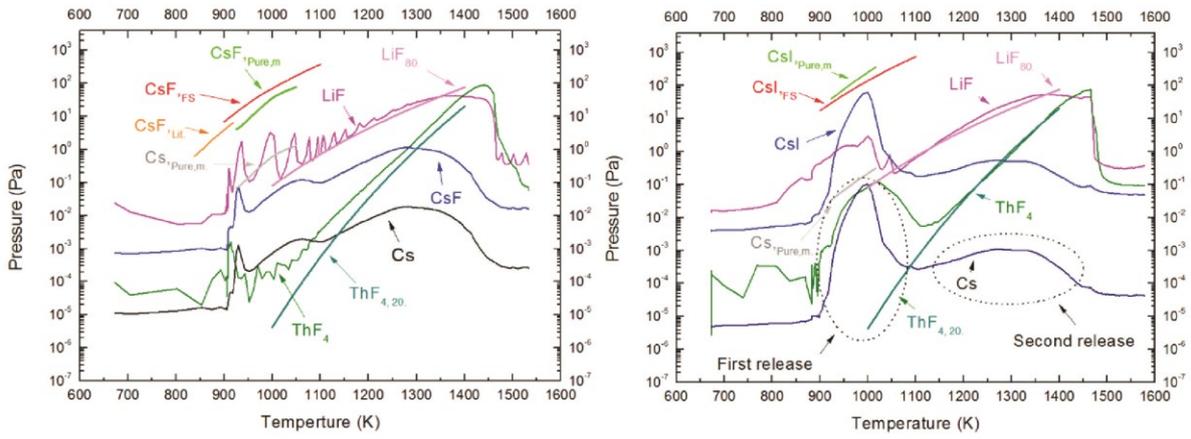
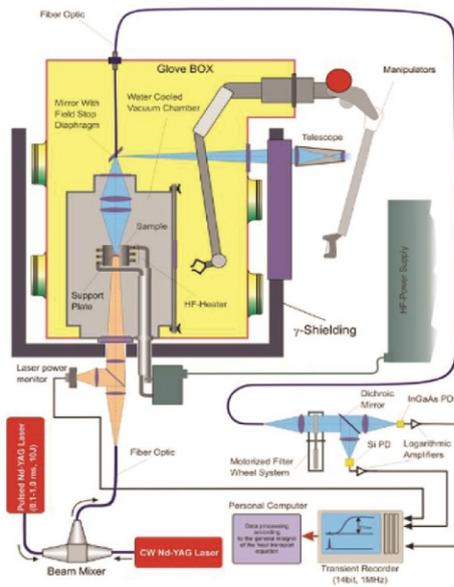


FIG. 40. Left: Volatility of CsF from the LiF-ThF4 eutectic. Right: Volatility of CsI from the LiF-ThF4 eutectic.

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(a) Schematic setup



(b) Photo of LAF

FIG. 41. Left: Schematic representation of the LAF device used at JRC for measurements of thermal conductivity; right: LAF device installed in the hot labs.

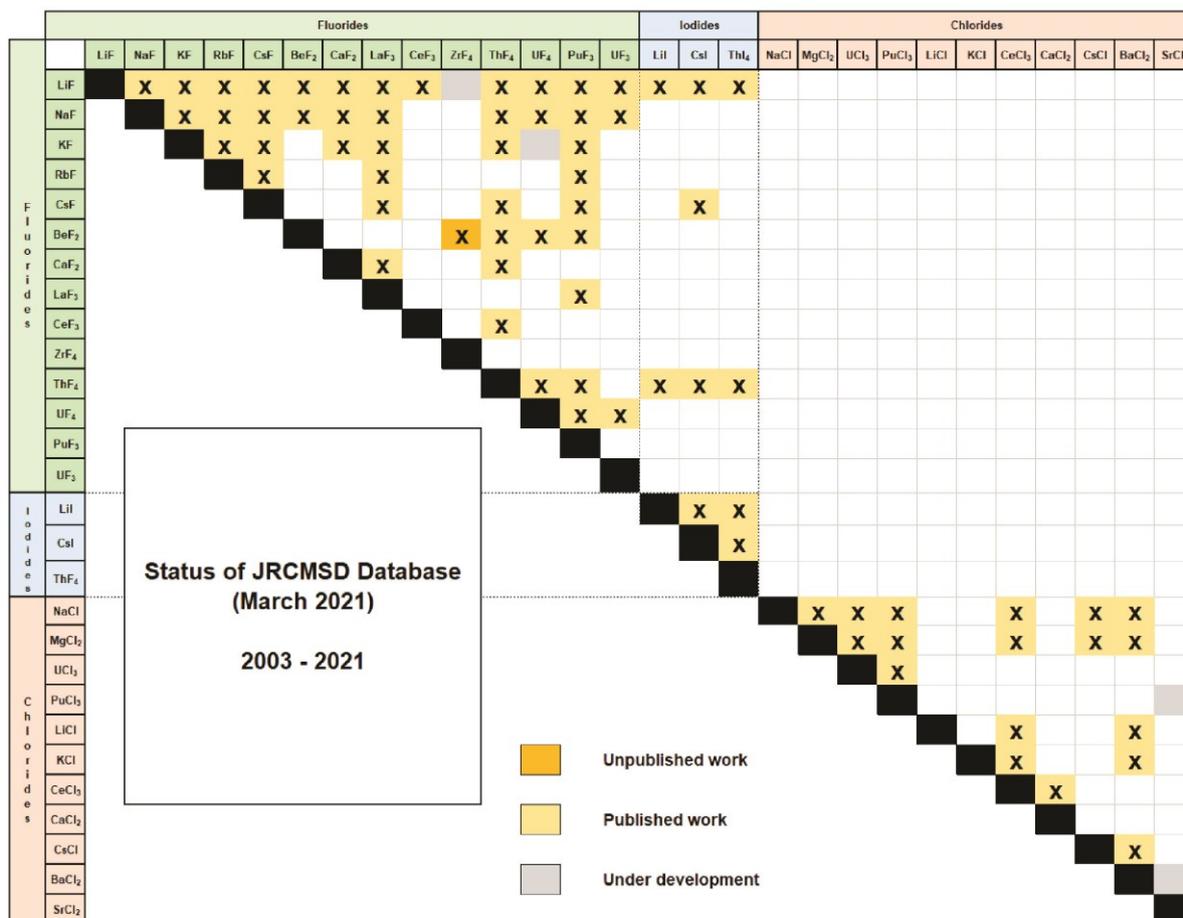


FIG. 42. Status of binary systems in the JRCMSD thermodynamic database.

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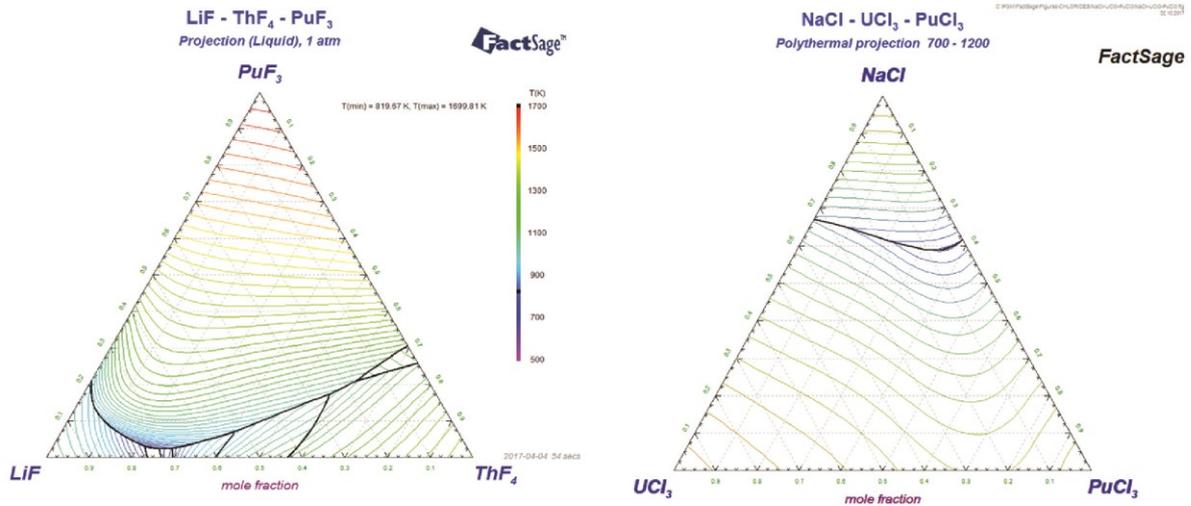


FIG. 43. The calculated liquidus projection of the LiF-ThF₄-PuF₃ and NaCl-UCl₃-PuCl₃ systems (left and right, respectively).

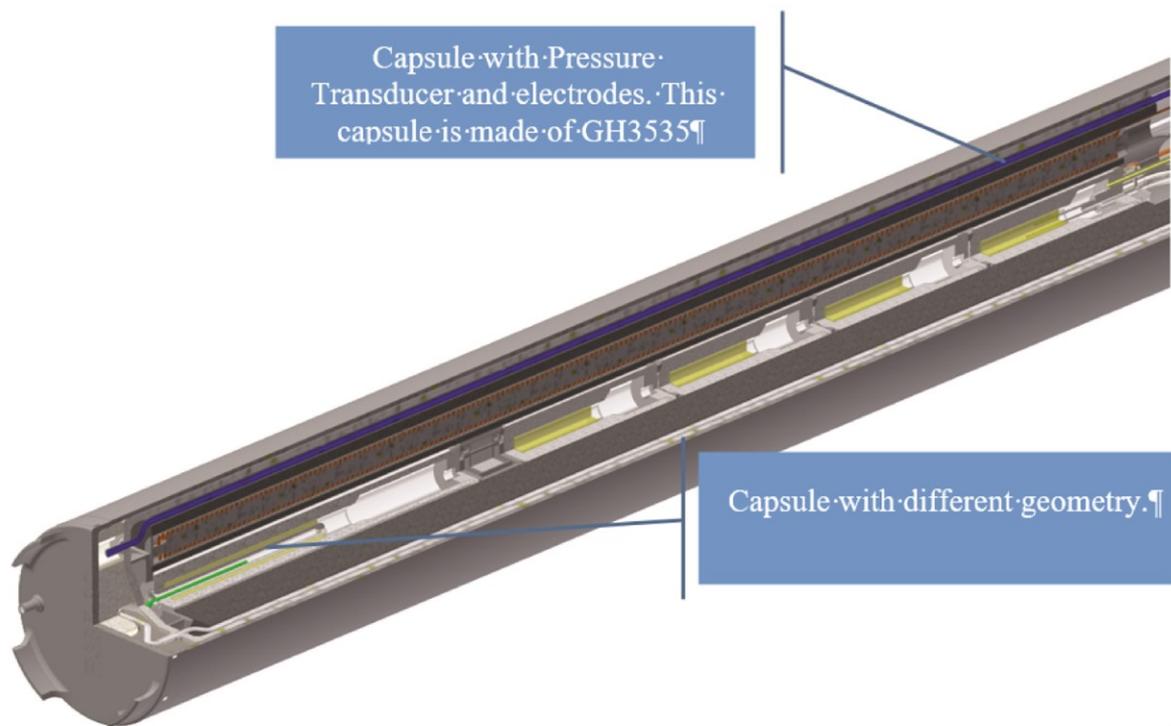


FIG. 44. Section of the SALIENT-03 experiment.

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Coupling LiCore-ALICES®
Toward a full simulator

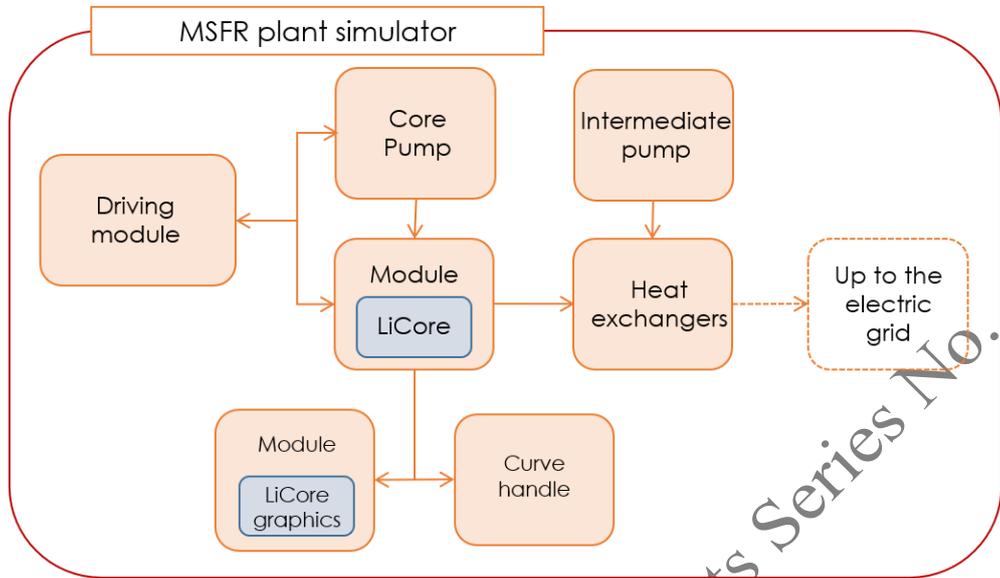


FIG. 45. Global scheme of the LiCore-ALICES power plant simulator.

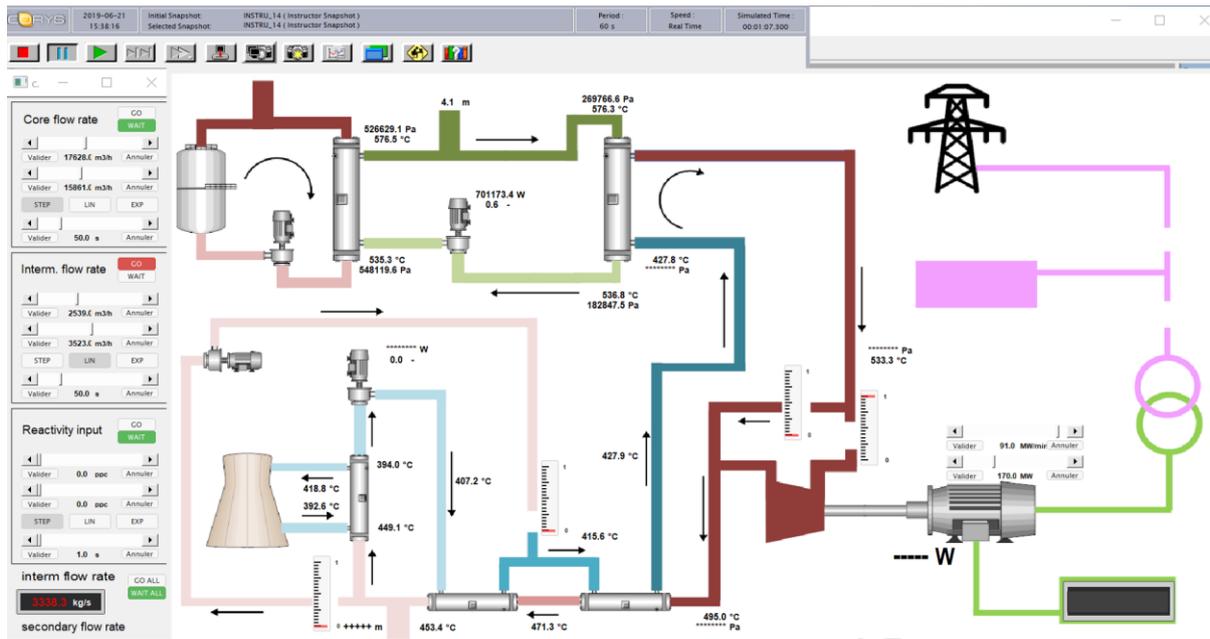
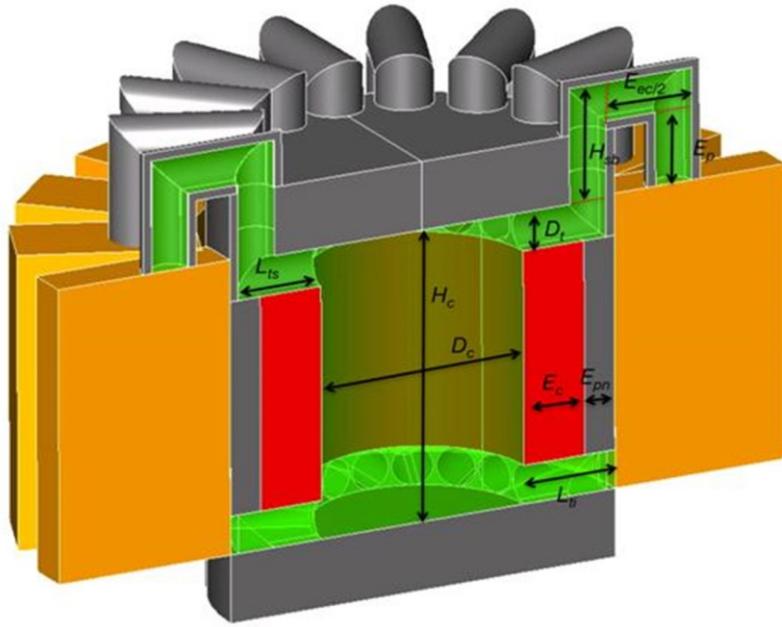


FIG. 46. Main screen of the LiCore-ALICES power plant simulator for the MSFR.



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FIG. 47. Sizing parameters for the fuel circuit of the MSFR.

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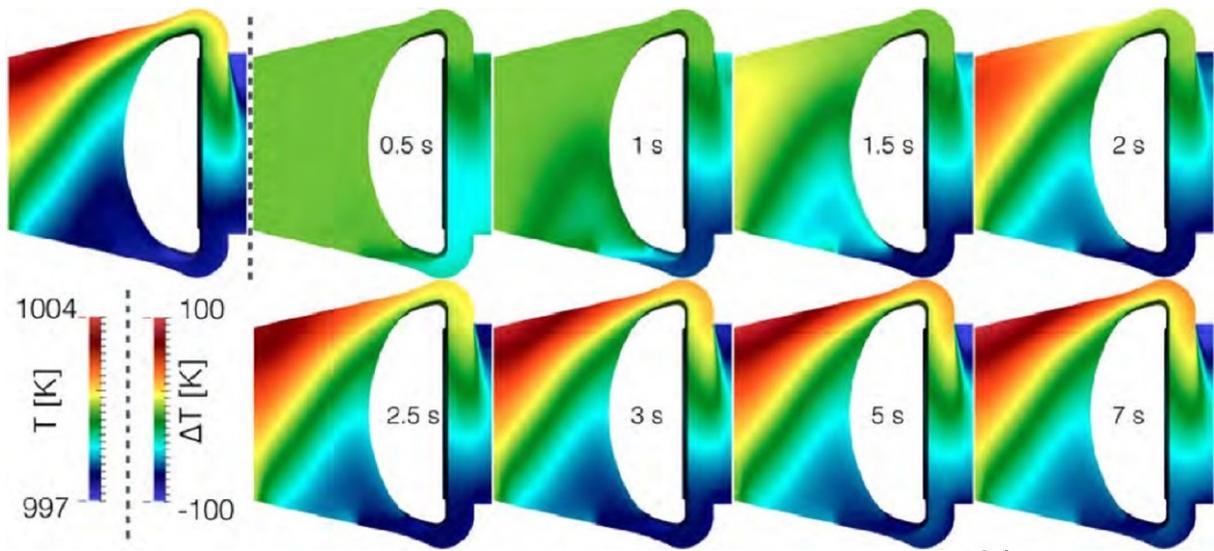


FIG. 48. Distribution of the fuel salt temperature at $t = 0$ (left), and its time variation $T(t) - T(0)$ for a 100 MW to 3 GW overcooling transient.

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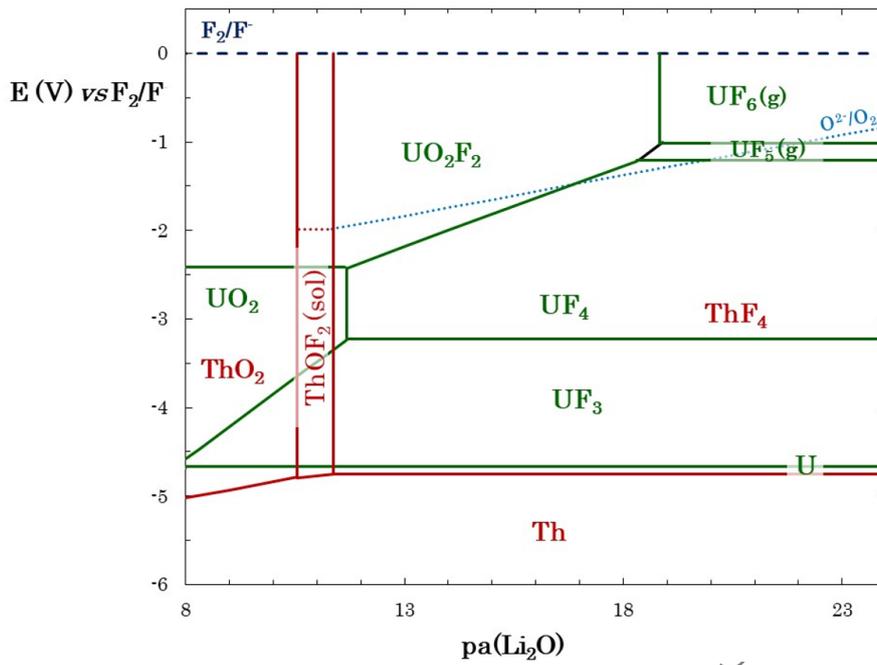
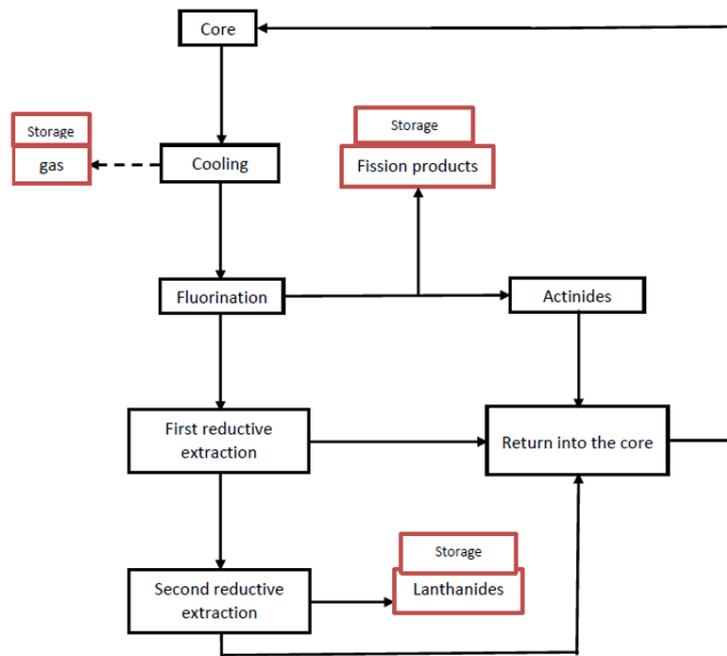


FIG. 49. Thermodynamic diagram of stability of uranium in LiF-ThF4 molten salt at a temperature of 650°C ($P(F_2) = 1 \text{ atm}$, $a(F^-) = 1$, $[UF_4] = 2.5 \text{ mol\%}$) [253].

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FIG. 50. Main steps of the chemical processing scheme of the MSFR fuel salt, including the physical (bubbling) processing on the left top (gas) [255]

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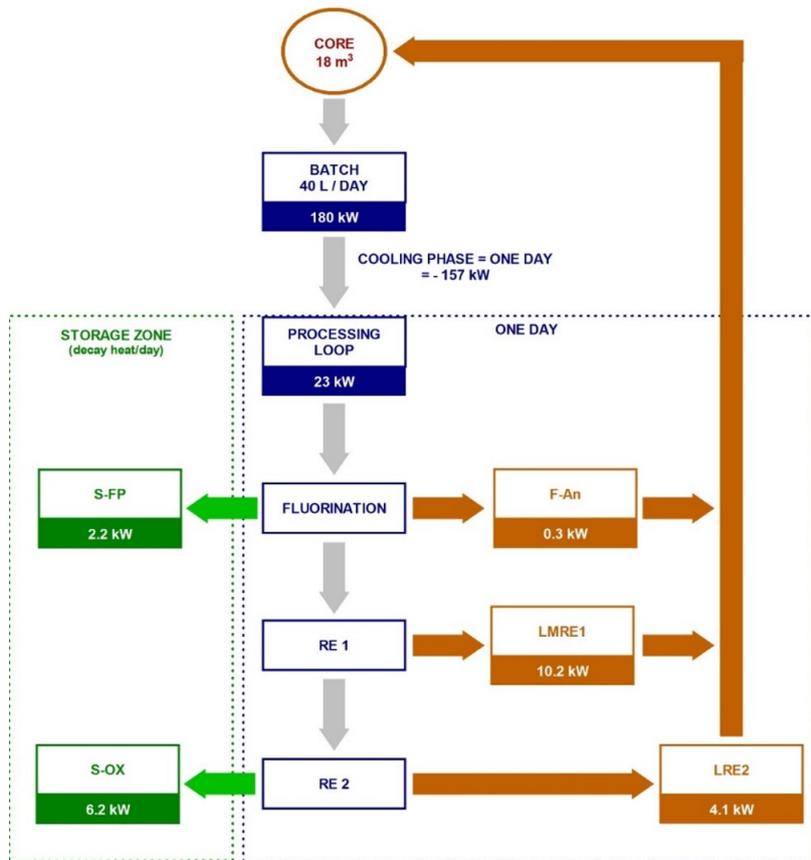


FIG. 51. Decay heat in each step of the processing scheme and in the storage zone.

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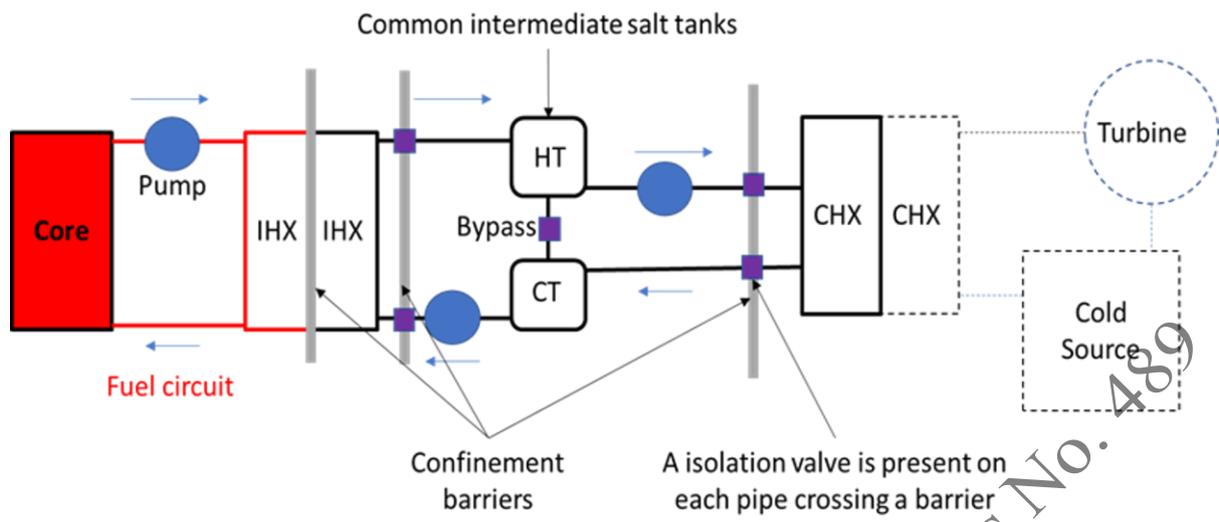


FIG. 52. Functional structure of the intermediate salt circuit of the MSFR.

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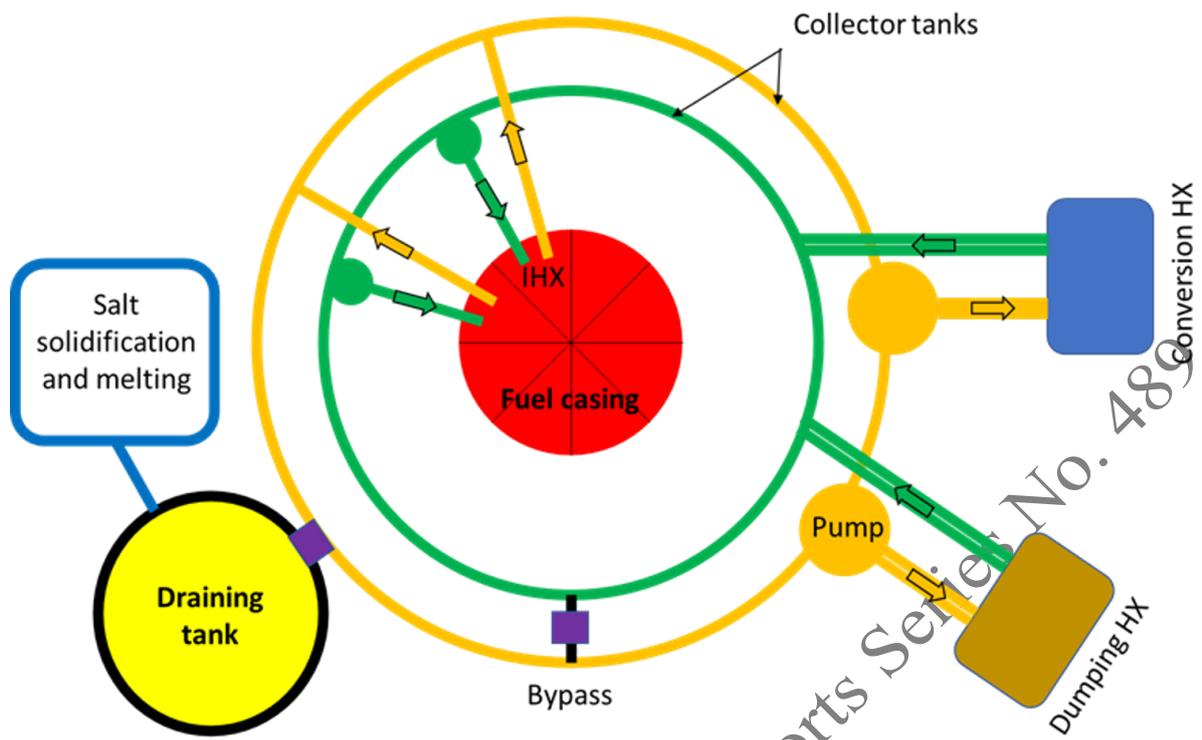


FIG. 53. Schematic presentation of the intermediate salt circuit of the MSFR.

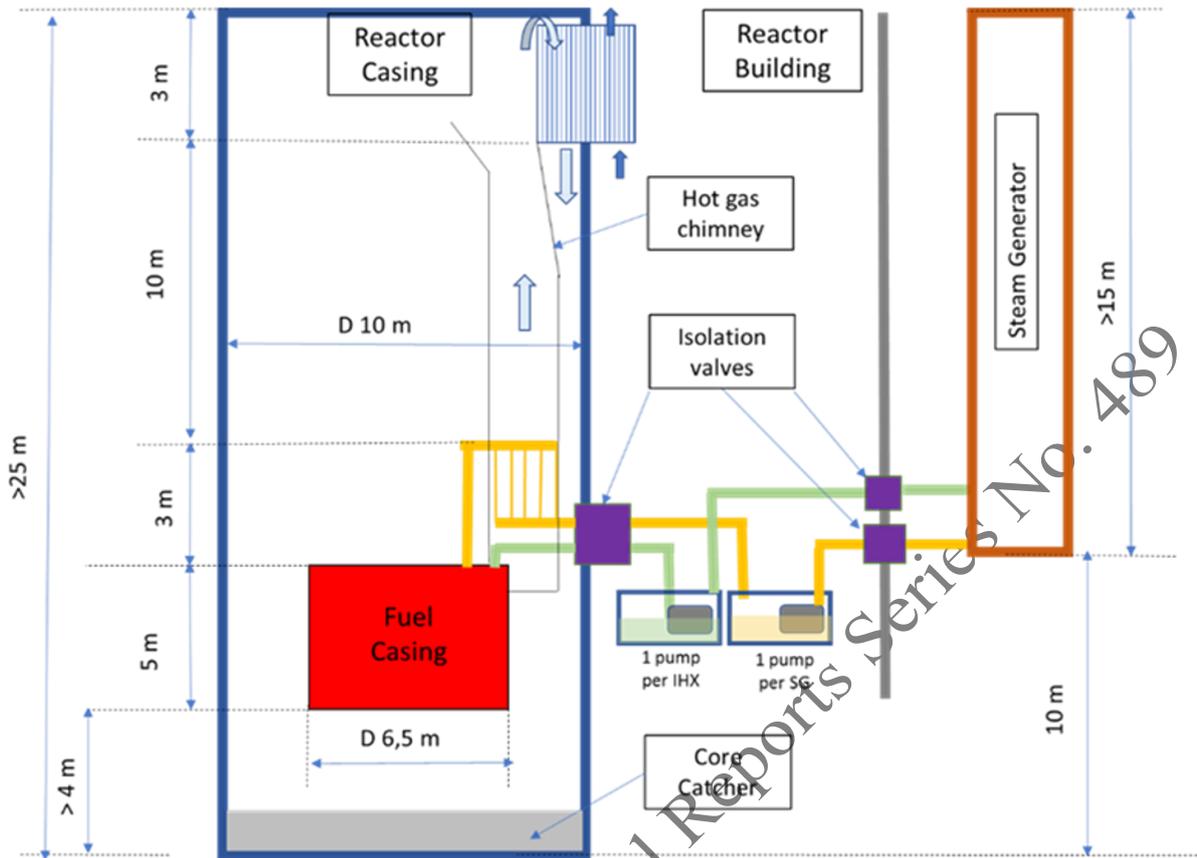


FIG. 54. Attempt of MSFR plant sizing in case of absence of emergency draining tank.

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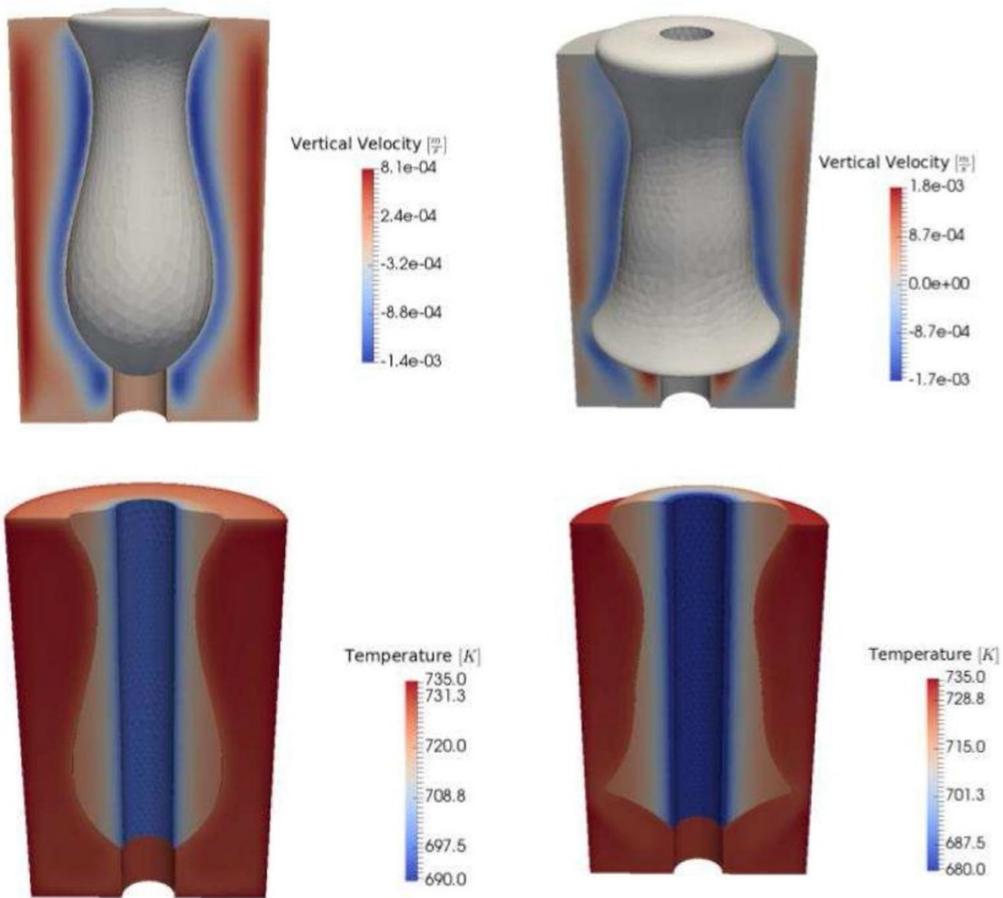


FIG. 55. Temperature (bottom) and flow (top) fields during the solidification process [266].

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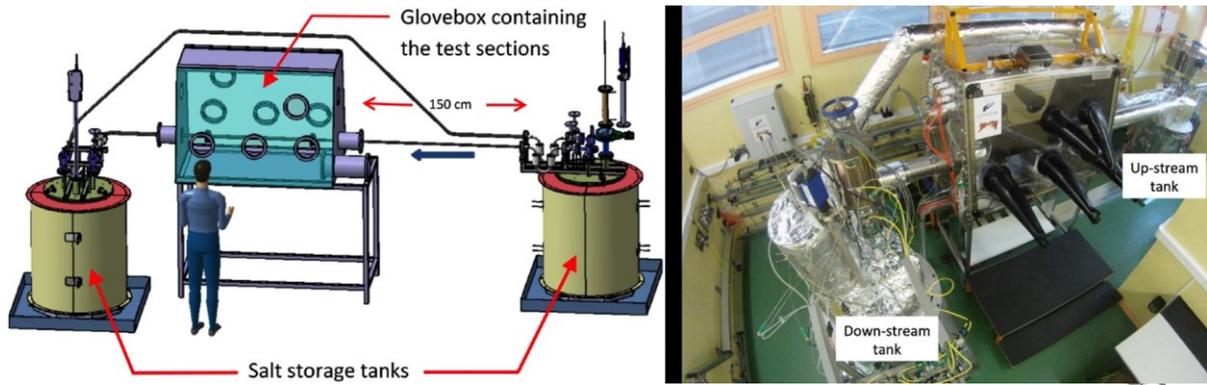


FIG. 56. Schematic (left) and photo (right) of the SWATH facility.

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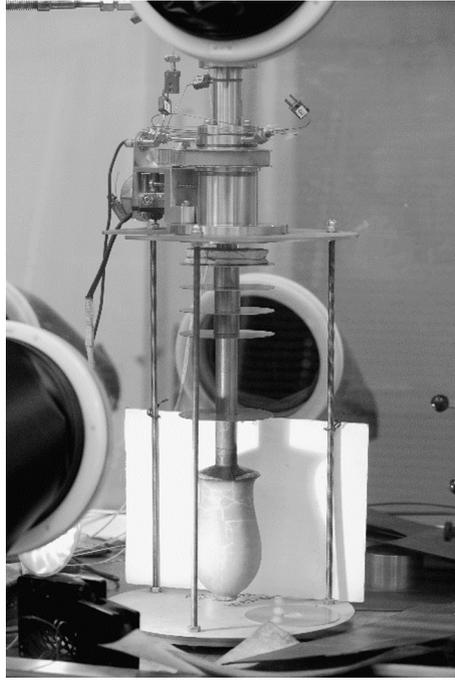


FIG. 57. Picture of the solidification experiment.

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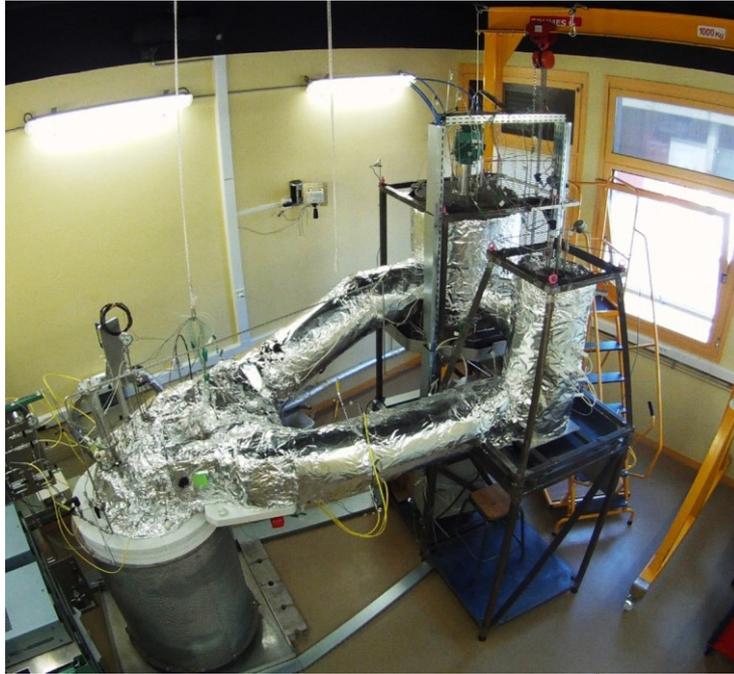


FIG. 58. View of the FFFER facility with thermal isolation around the pipes.

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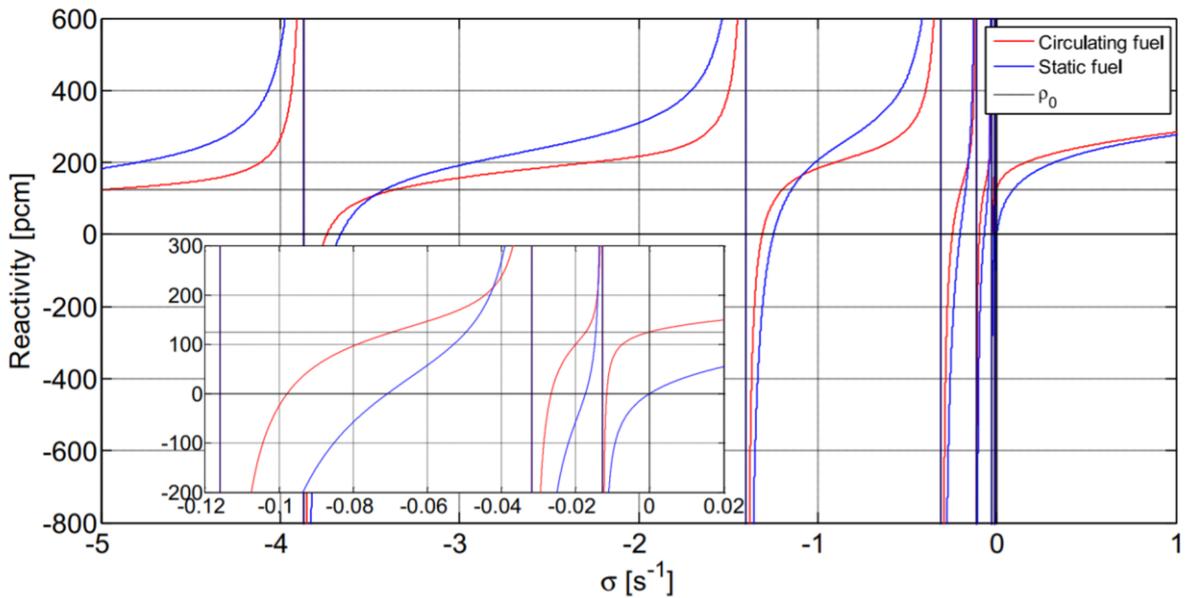
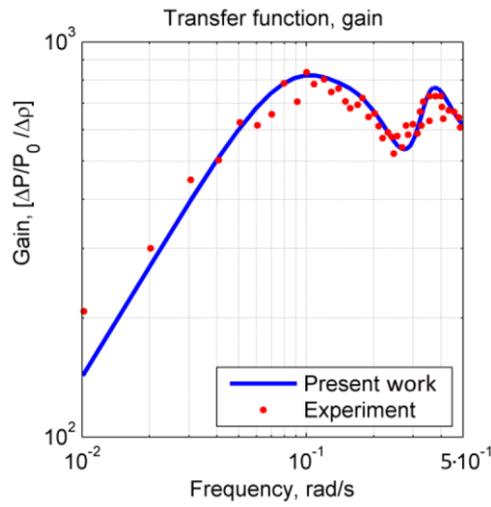


FIG. 59. Top: Gain of the transfer function between power variation ($\Delta P/P_0$, where P_0 is the nominal power) and reactivity ($\Delta\rho$) for the MSRE at 8 MW (blue solid line – “Present work”) and comparison with experimental data (red dots – “Experiment”) [273]. Bottom: Inhour equation for the MSBR with circulating and static fuel (ρ_0 being reactivity compensation for the loss of delayed neutrons and σ being the reciprocal time constant) [275].

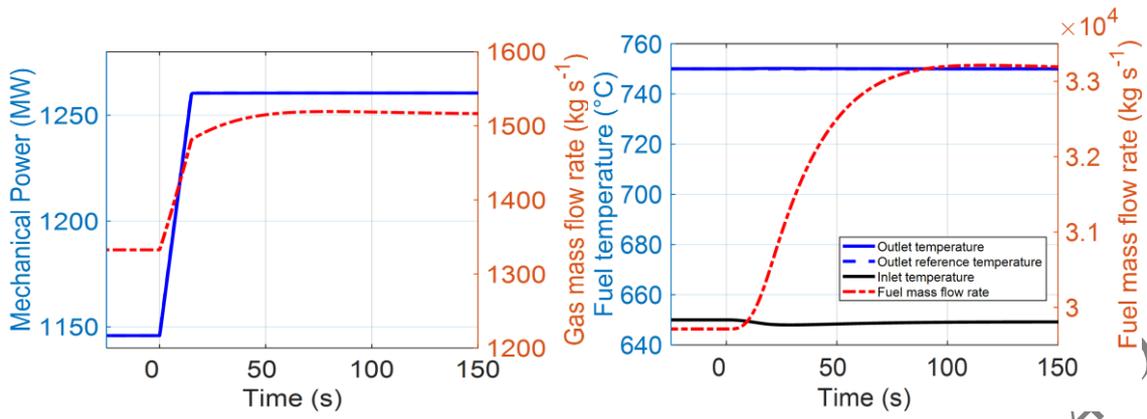


FIG. 60. Transient response of the control system for the MSFR without employing external reactivity (control strategy of mass flow rates). Left: Mechanical power and gas flow rate. Right: fuel temperatures and fuel mass flow rate.

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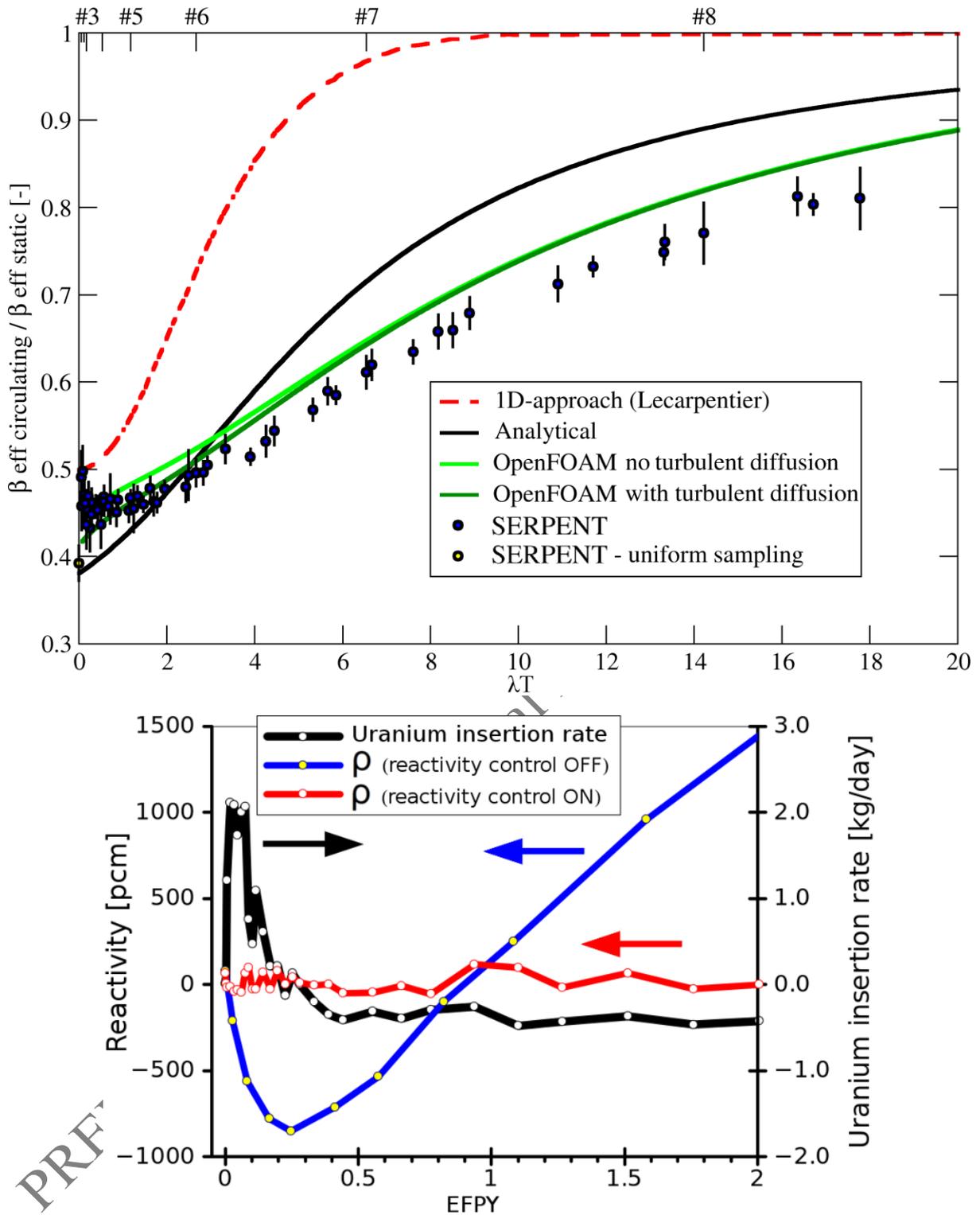
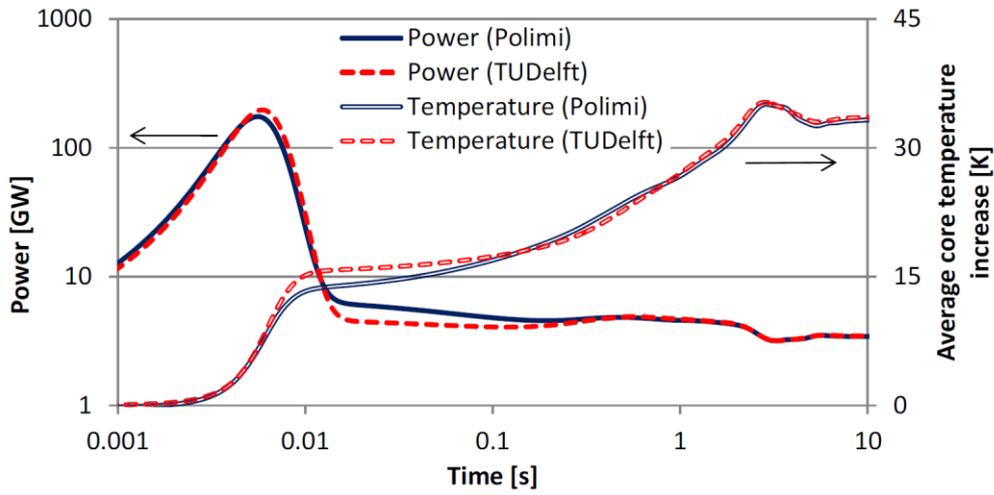


FIG. 61. Top: β_{eff} for the MSFR: comparison between analytical, based on Monte Carlo (SERPENT) and deterministic approaches (OpenFOAM) as function of the product between the precursor decay constant λ and the fuel salt loop circulation period T [279]. Bottom: Reactivity and uranium insertion rate for the transuranic-started version of the MSFR obtained with the reactivity control algorithm [280].



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Time: 3.39s

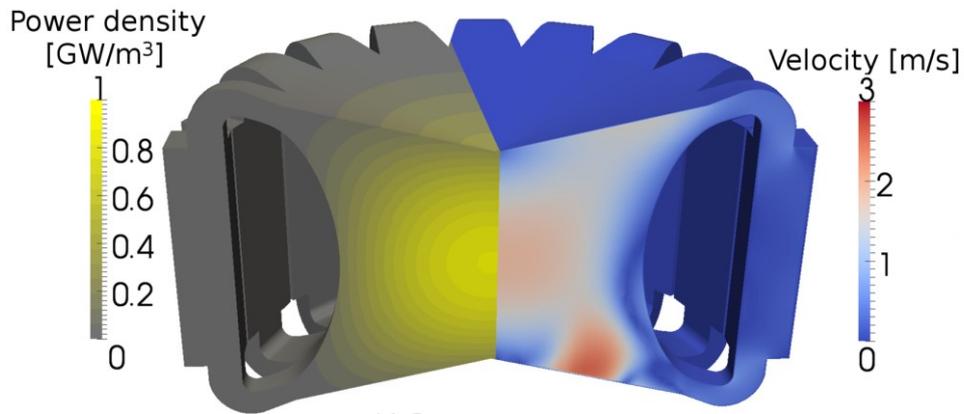


FIG. 62. Top: Evolution of power and average core temperature in the MSFR core after a stepwise reactivity insertion equal to 200 pcm (super-prompt-critical) with a 2D axial-symmetric model [283]. Bottom: Velocity and power density fields in the MSFR after an accident of a single pump failure with a 3D model [180].

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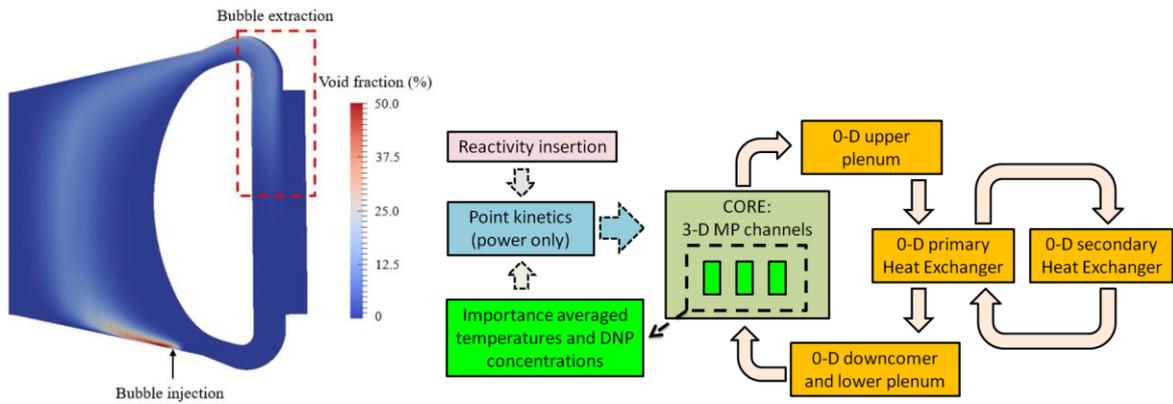


FIG. 63. Left: Bubble distribution (void fraction) in the MSFR obtained with the two-phase compressible solver [286]. Right: Conceptual scheme of the Geometric Multiscale model of the MSRE [273]; DNP is Decay Neutron Precursors, and MP is Multi-Physics.

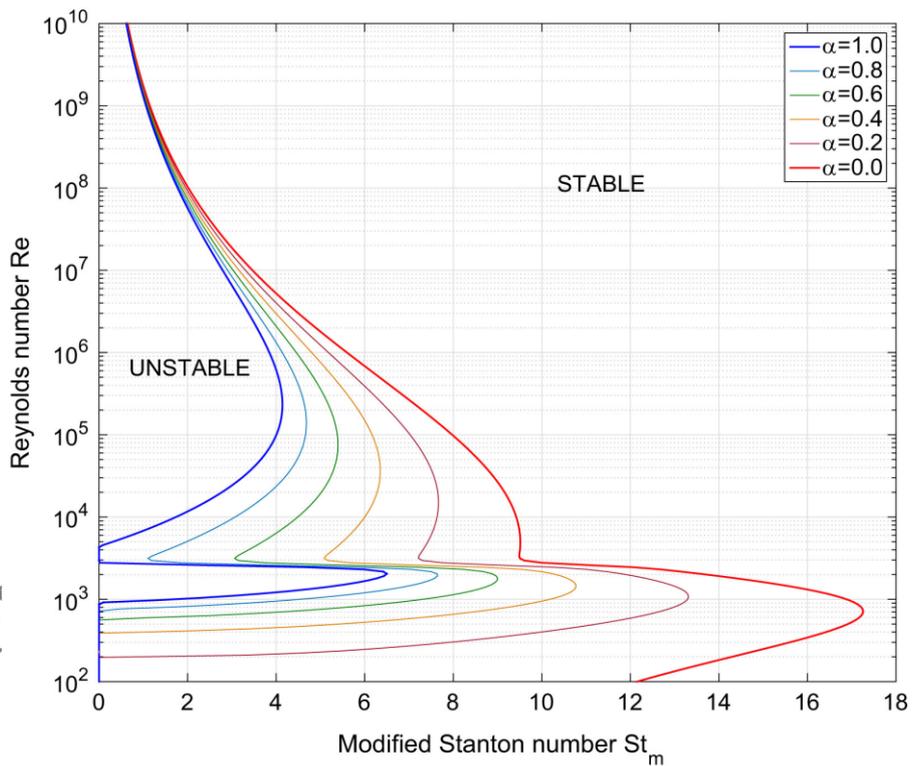
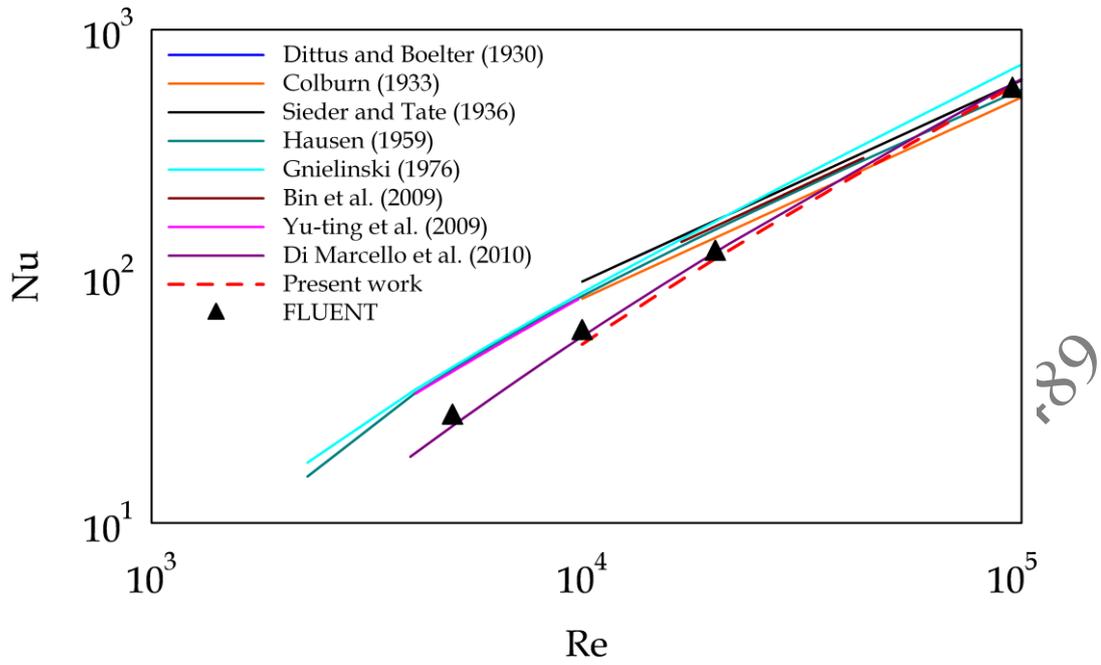


FIG. 64. Top: Comparison among heat transfer correlations and CFD results obtained by means of FLUENT software, in the Reynolds range of interest for MSRs ($Pr = 11$) [271]. Bottom: Stability map (modal analysis) of a natural circulation loop for various levels of internal heat generation, from $\alpha = 1$ (conventional, localized heat flux only) to $\alpha = 0$ (internal heat generation only) [294].

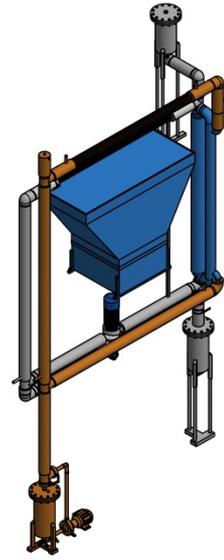
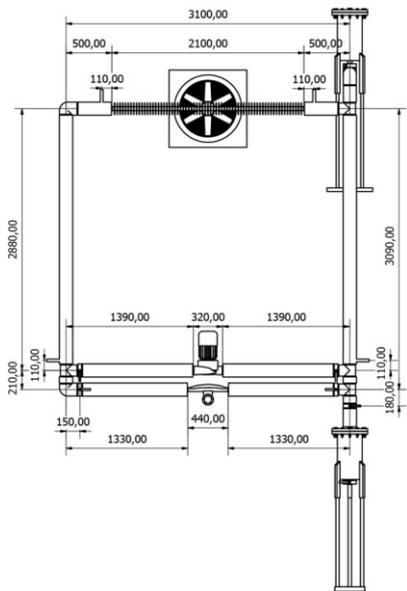


FIG. 65. Left and centre: The DYNASTY facility built at POLIMI Labs. Right: in orange, the extension of the facility (e-DYNASTY) to study coupled natural circulation loops.

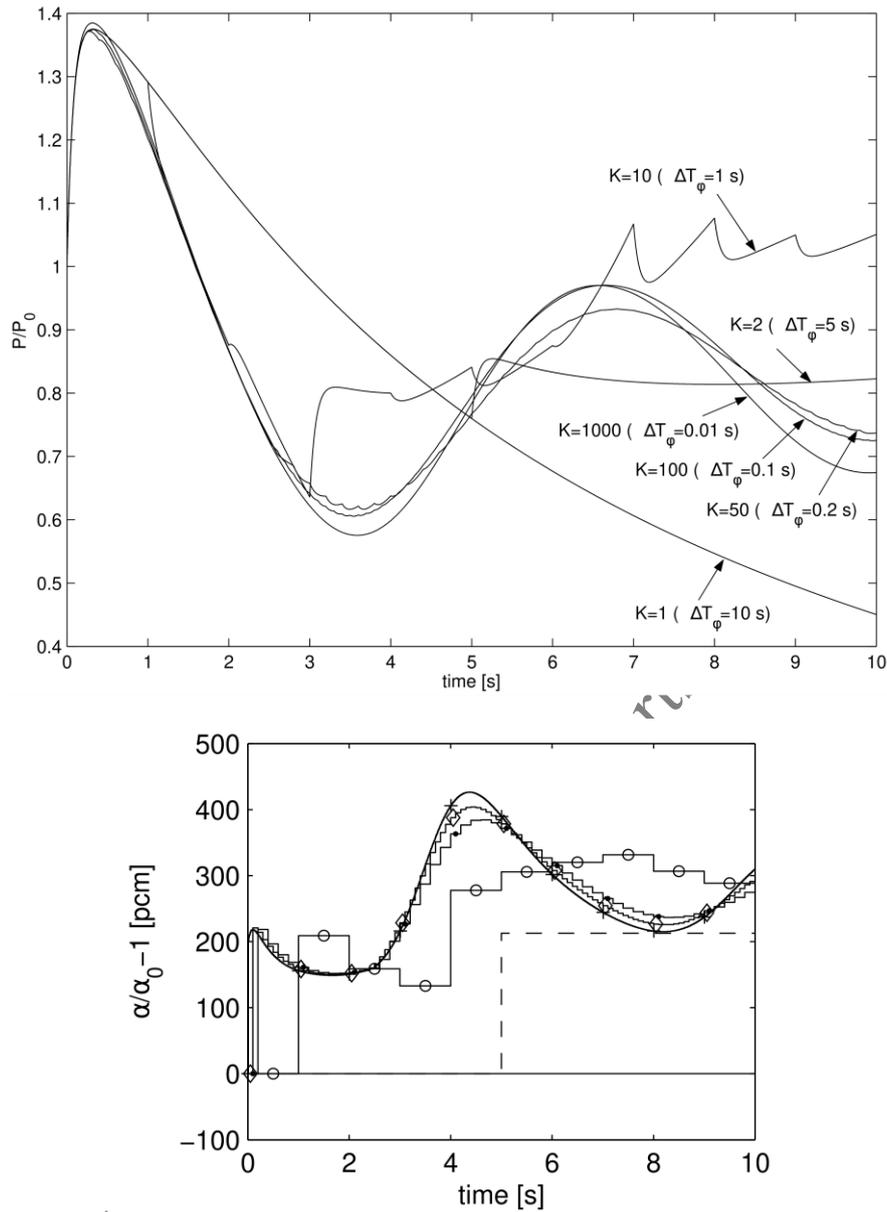


FIG. 66. Top: Power evolution, normalized to initial value, for different numbers K of shape recalculations for a transient of fluid velocity in a critical system (ΔT_ϕ is the time interval adopted for the shape recalculation). Bottom: Time behaviour of α (kinetic parameter of point kinetics related to reactivity evolution, see detail in Ref. [301]), normalized to its initial value α_0 , for different numbers of shape recalculations (solid line: $K = 1$; dashed line: $K = 2$; circles: $K = 10$; dots: $K = 50$; diamonds: $K = 100$; plus sign: $K = 1000$).

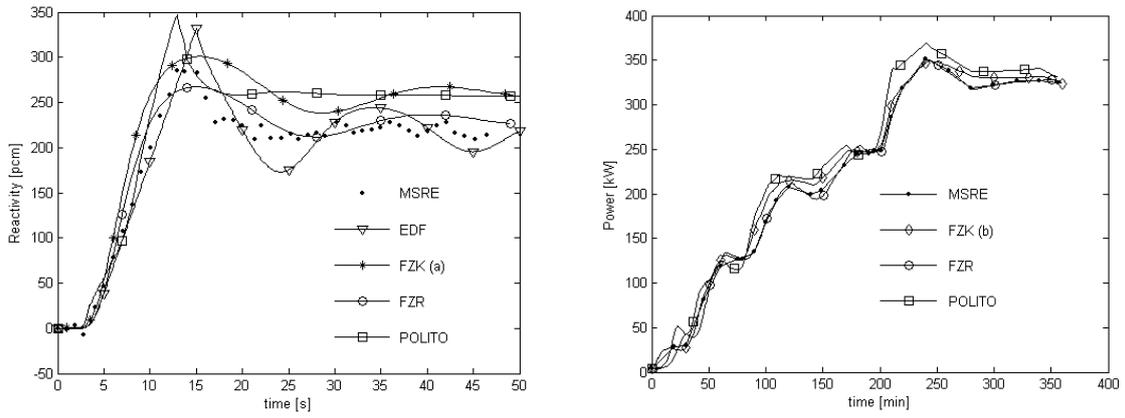


FIG. 67. Pump start-up transient in MSRE (JEF decay data) (left) and Natural convection transient in MSRE (right) [303].

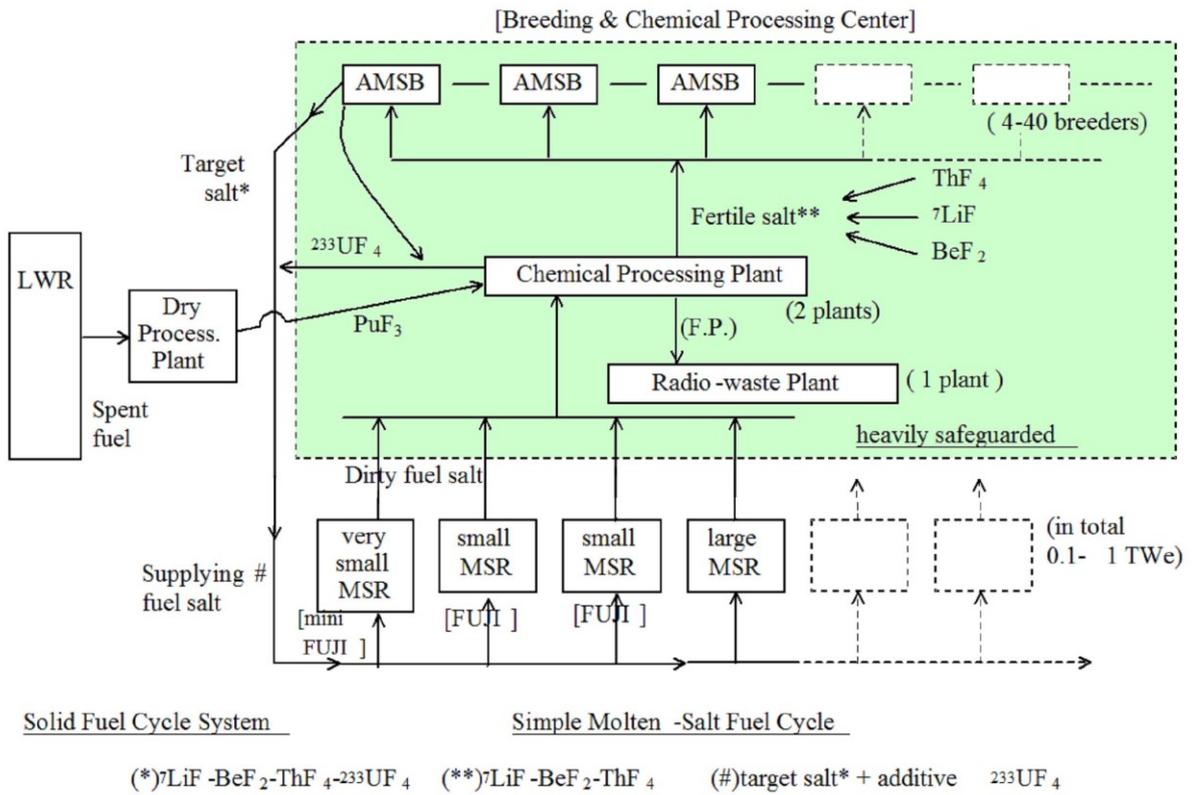


FIG. 68. Thorium Molten Salt Nuclear Energy Synergetic System (THORIMS-NES).

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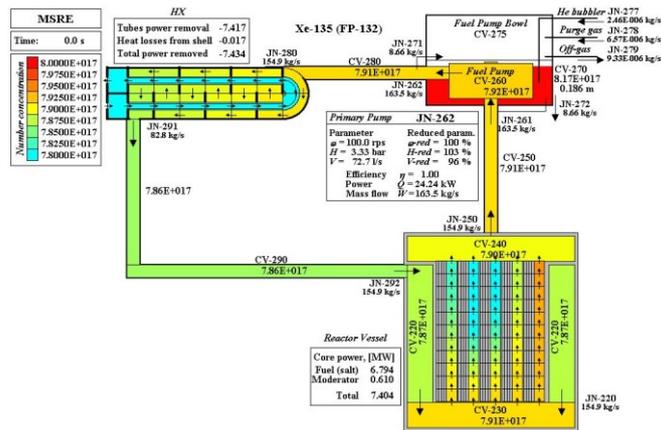


FIG. 69. Steady state distribution of noble gases in MSRE using the SPECTRA code.

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FIG. 70. Overview over the HFR.

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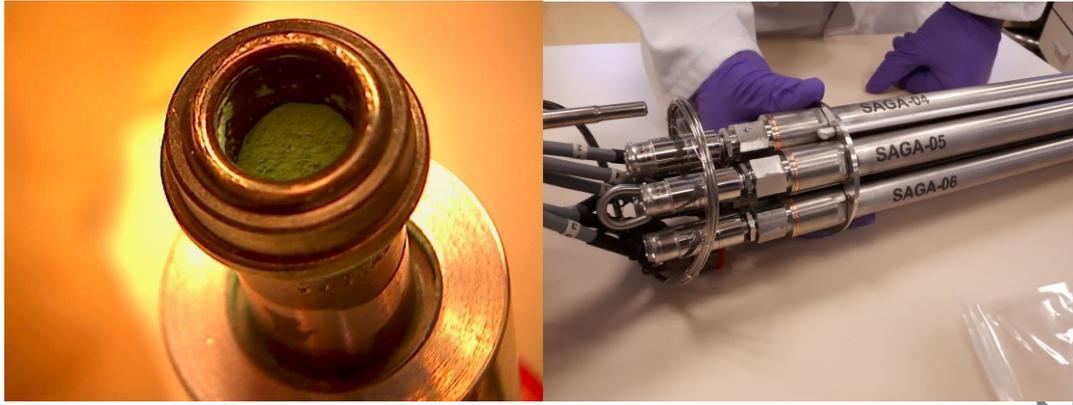


FIG. 71. Left: one of the salt-filled SAGA capsules before sealing. Right: the completed assembly of 6 capsules before loading into the SAGA facility.

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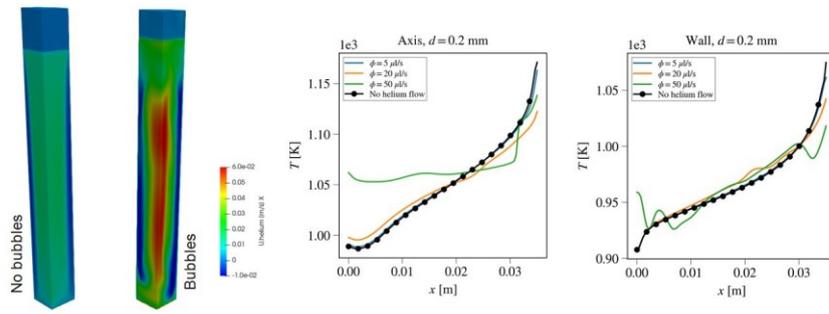


FIG. 72. Left: salt velocity profile. Center and right: axial temperature profiles for different flow rates of bubbles of diameter 0.2 mm (Center: profile at the centre of the salt column; right: profile at the salt-capsule interface).

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FIG. 73. Left: A segment of MONICR plate as supplied by CV Rez. Right: tensile samples fabricated at CV Rez from the same material.

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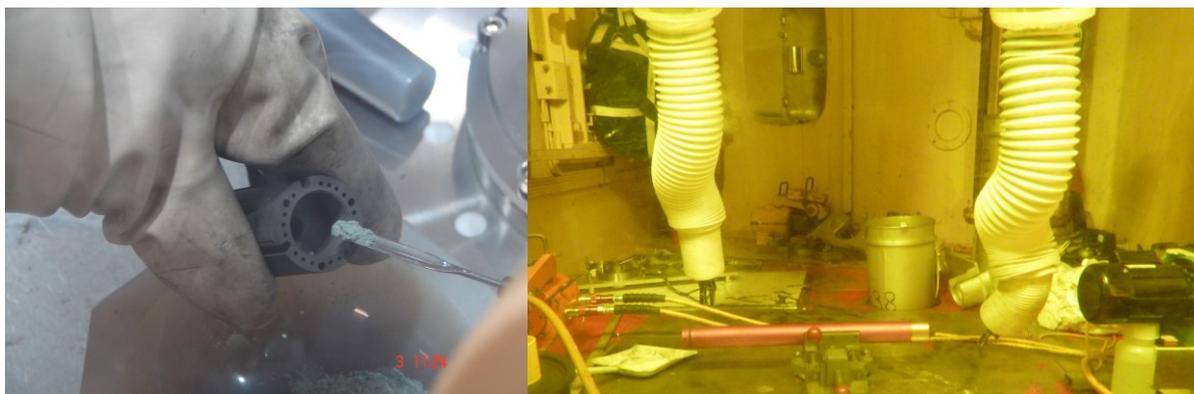


FIG. 75. Left: Loading of the LiF-ThF₄ salt samples in an inert glovebox environment at JRC Karlsruhe. Right: The SALIENT-01 sample holder after irradiation. Following cutting of the instrument lines in the DM-cell of the HFR, the sample holder is inserted into a gastight holder for transport to the Hot Cell Laboratories.

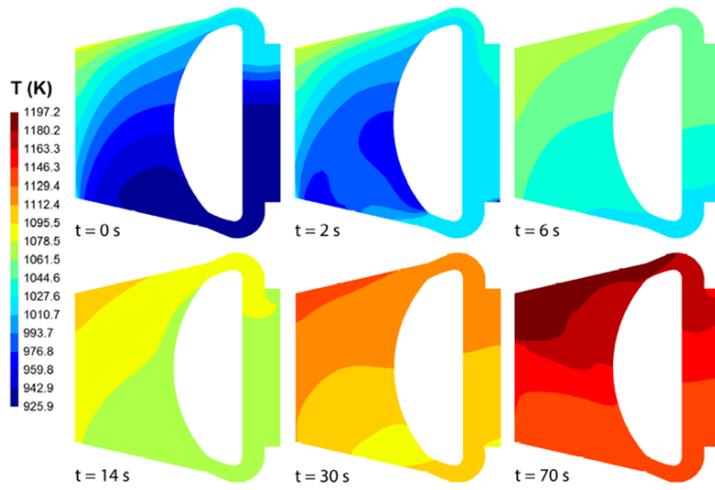


FIG. 76. Evolution of the fuel salt's temperature distribution during a TLOP transient [344].

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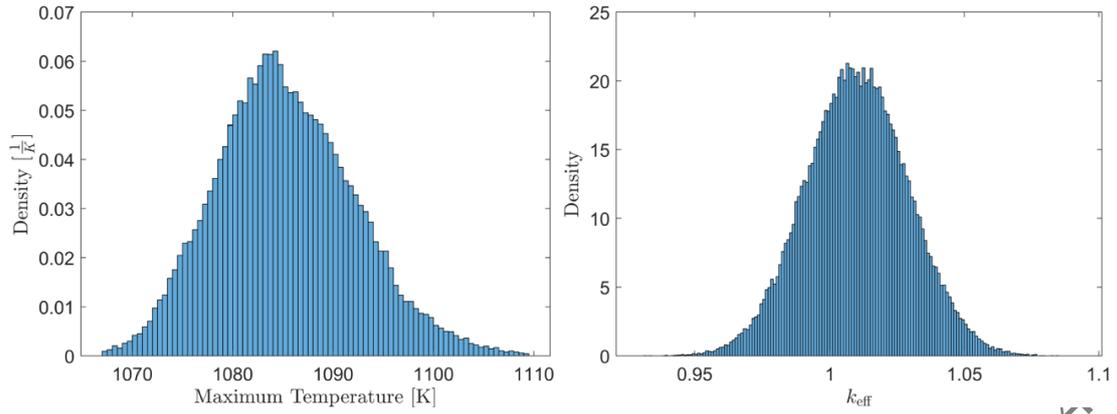


FIG. 77. Probability distributions of the maximum salt temperature in the core and of the multiplication factor, k_{eff} , under influence of input parameter variation.

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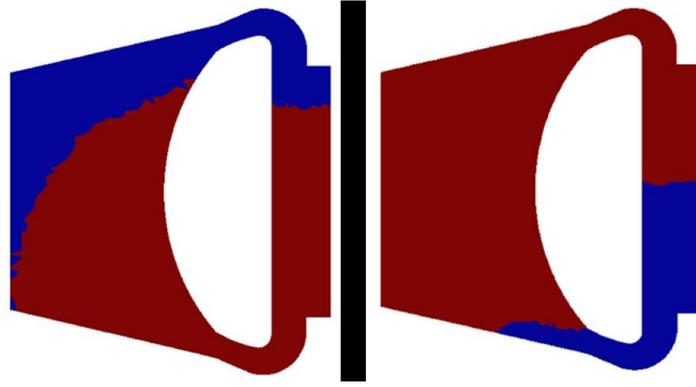


FIG. 78. Left: Domain where $T < 1023^{\circ}\text{K}$ with probability higher (red) and lower (blue) than 95%. Right: Domain where $T > 923^{\circ}\text{K}$ with probability higher (red) and lower (blue) than 95%.

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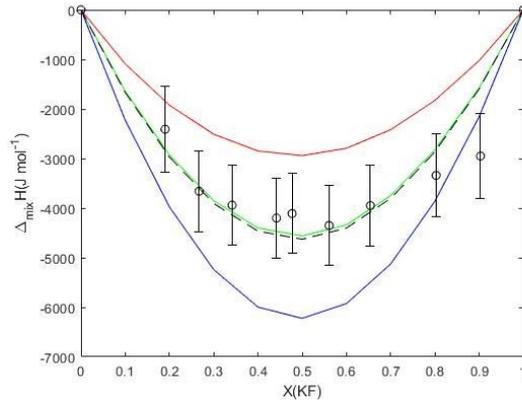
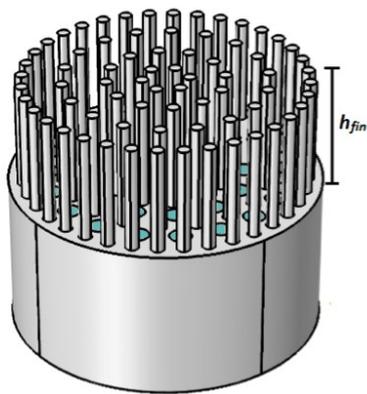
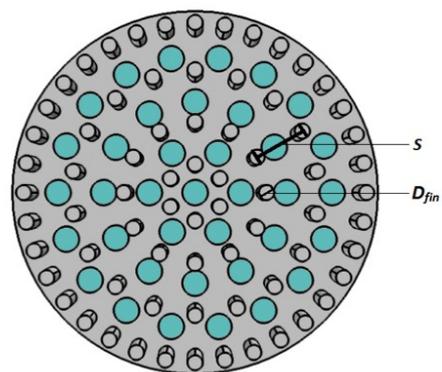


FIG. 79. Comparison of measured and CALPHAD simulated mixing enthalpy values together with derived uncertainties. Red, green, and blue lines represent the 0th, 50th and 100th percentiles of the calculated phase diagram, while the black circles showcase the measured values together with their experimental uncertainties. The black dotted line corresponds to the mixing enthalpy computed using the thermodynamic model implemented in the JRCMSD database (see Subsection 5.10.2.12).

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Design including cooling fins.



Top view of design including cooling fins.

FIG. 80. Preliminary design of a freeze plug for the MSFR. Fins are installed to enhance the melting of the plugs that are present in the distribution plate.

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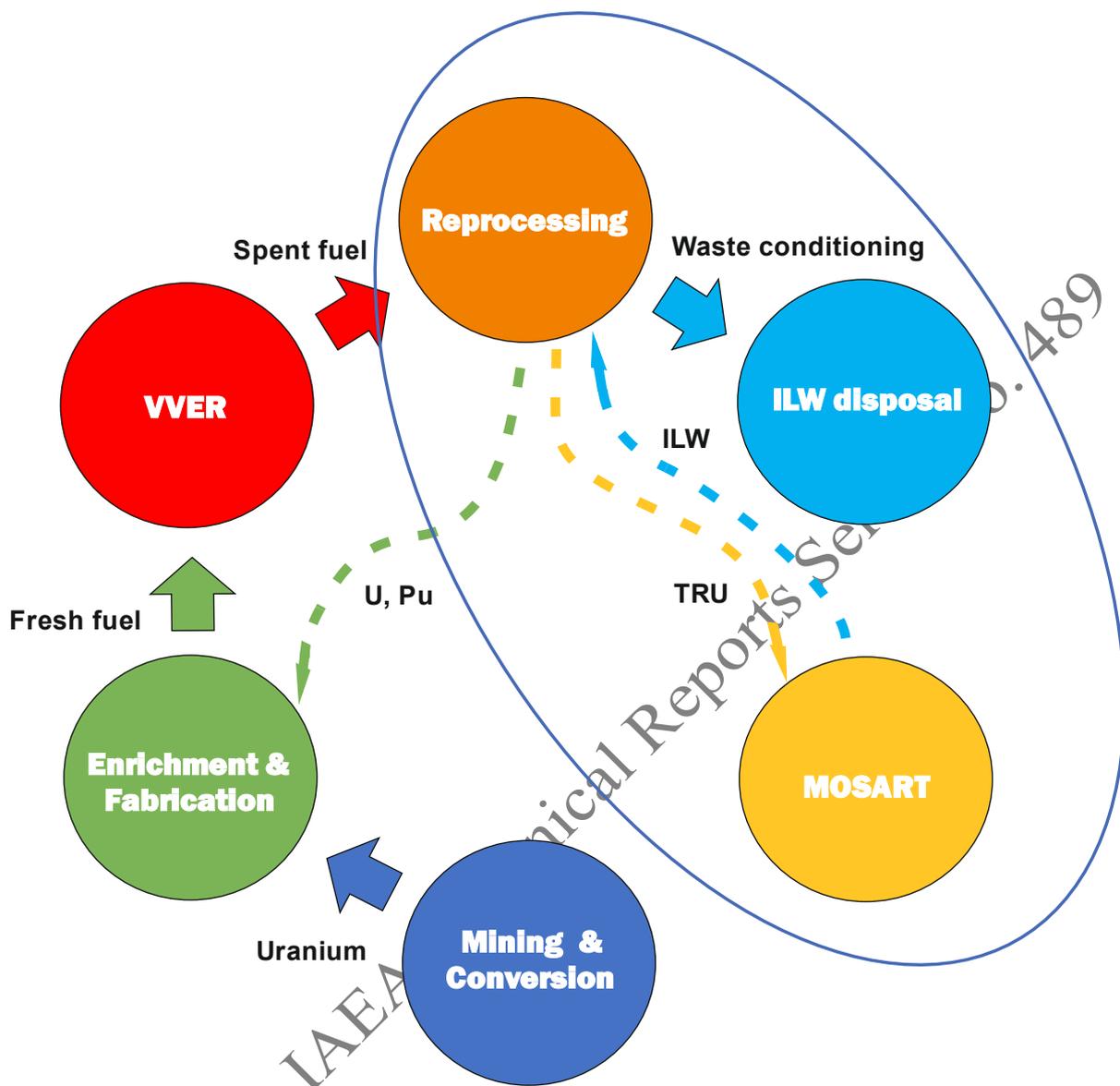


FIG. 81. Nuclear fuel cycle with MOSART at MCC site (ILW means intermediate level waste).

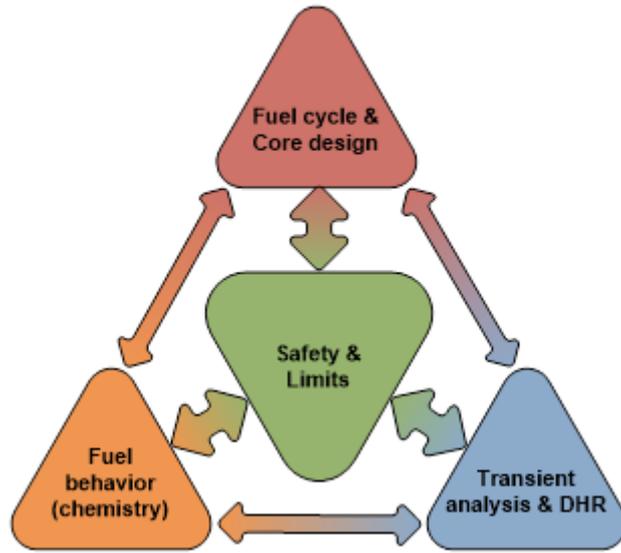


FIG. 83. Illustration of four research areas and their interdependencies.

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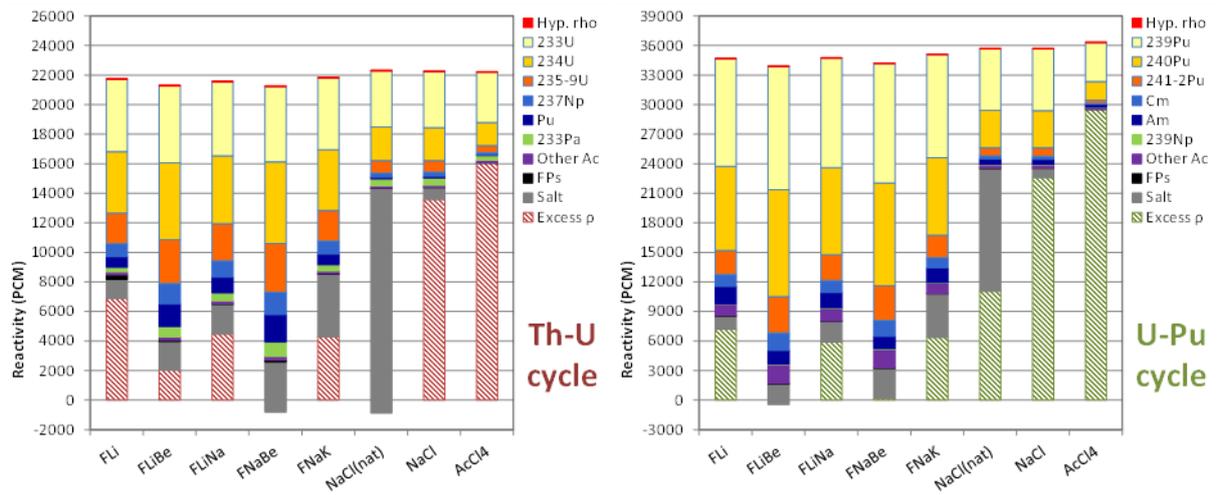


FIG. 84. Comparison of eight fast MSR designs using excess and hypothetical (for zero parasitic captures) reactivities with Th-U cycle (left) and U-Pu cycle (right).

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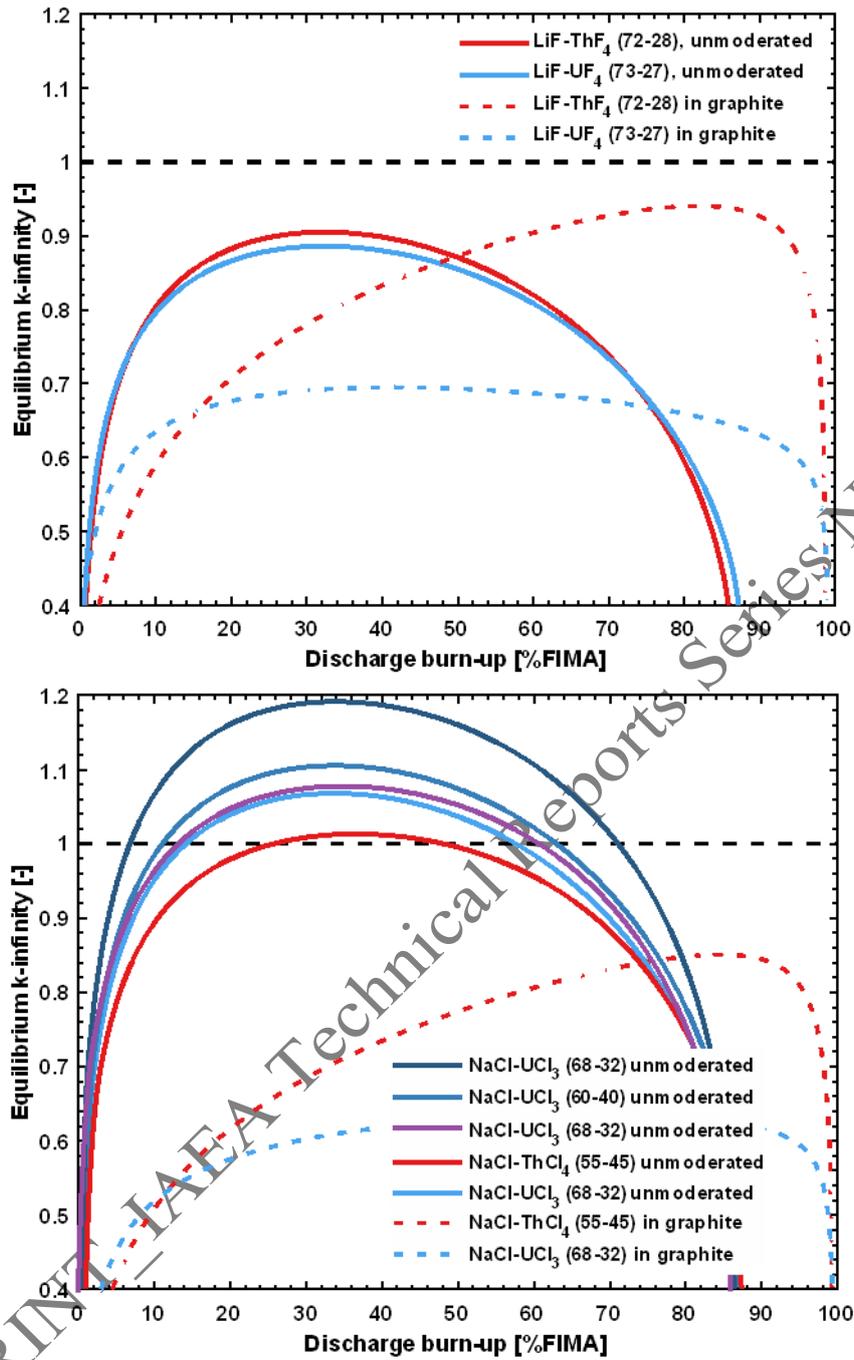


FIG. 85. Evolution of infinite multiplication factor in a breed-and-burn reactor as a function of fuel discharge burnup [127].

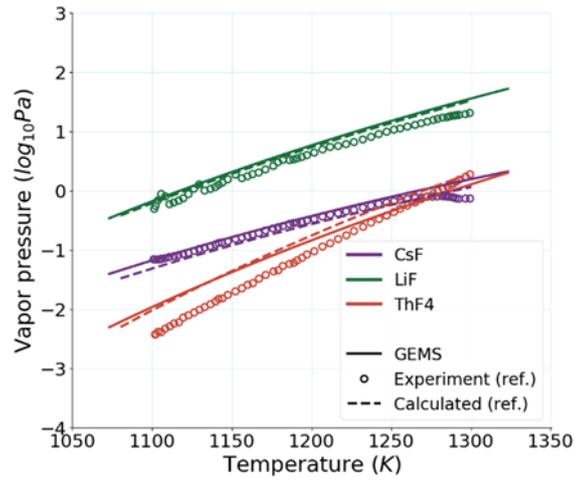


FIG. 86. Vapor pressure data for the LiF–ThF₄–CsF system calculated at PSI with GEMS and compared Ref. [221].

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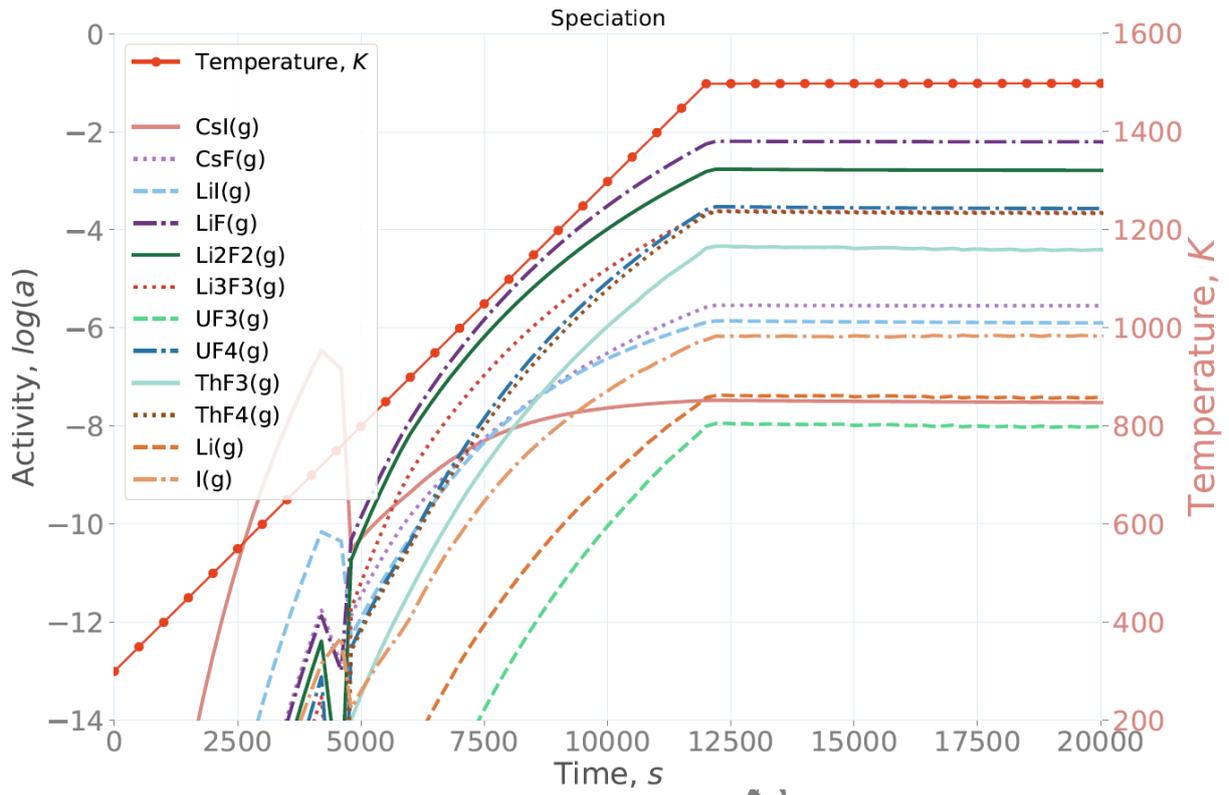


FIG. 112. Vapor pressures for the LiF–ThF₄–UF₄–Cs–I system calculated with GEMS.

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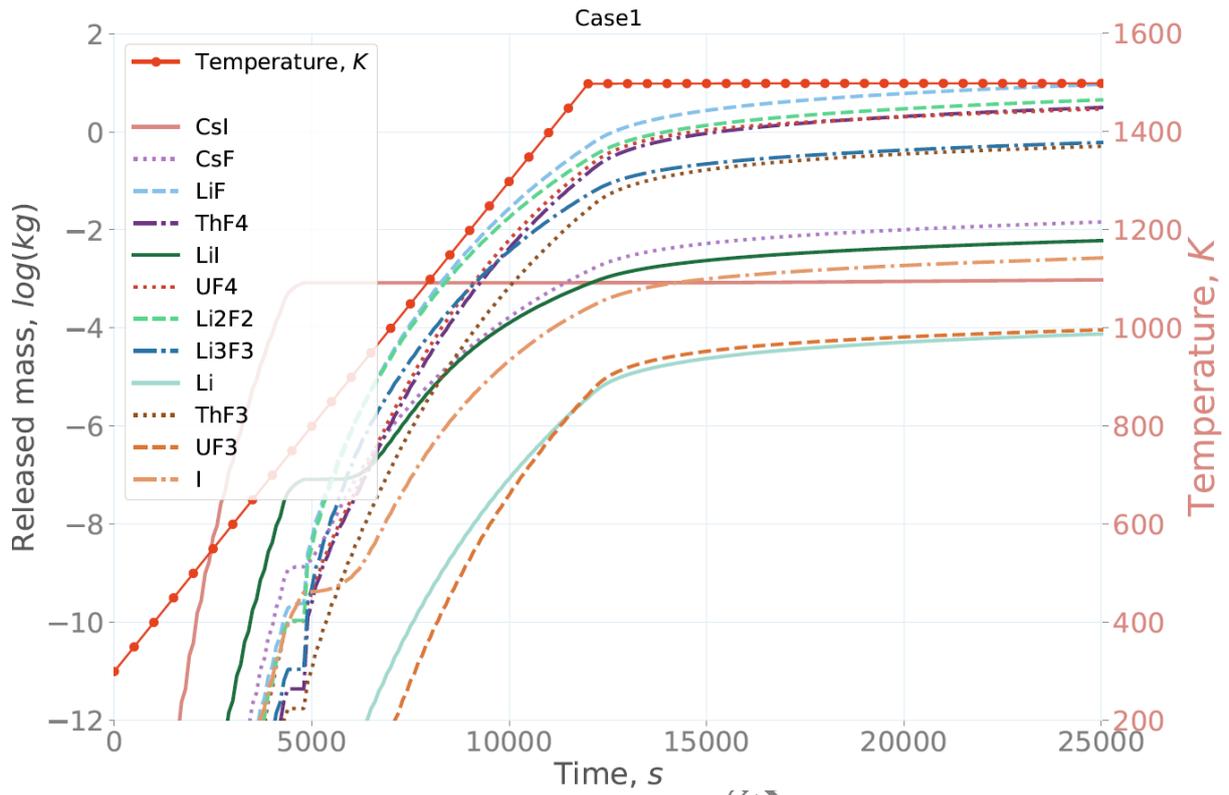


FIG. 88. Fission product release from molten salt pool during the heating process.

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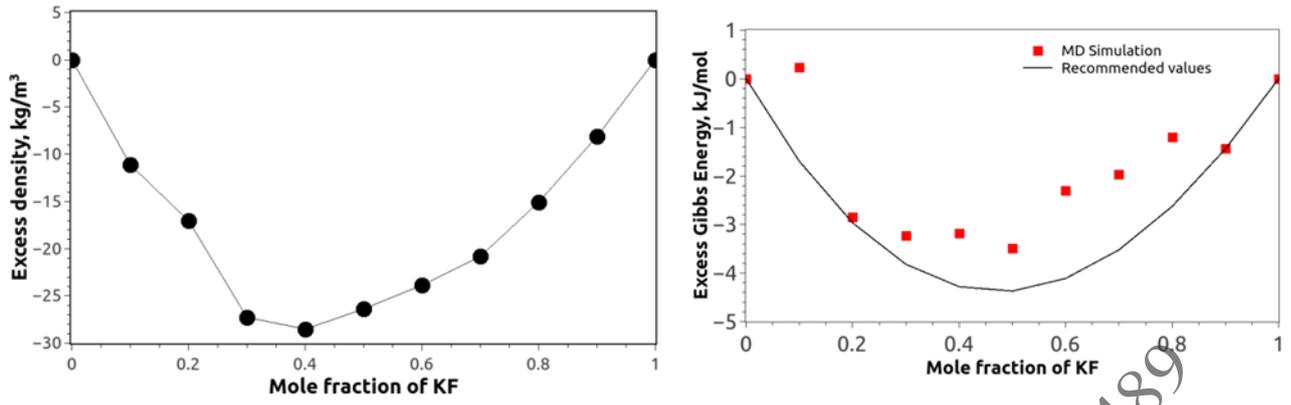


FIG. 89. Excess density (left) and excess Gibbs energy (right) of the Li-K-F system at 1,100°K. Left: The dots show the calculated values, and the line is a linear interpolation between two consecutive dots.

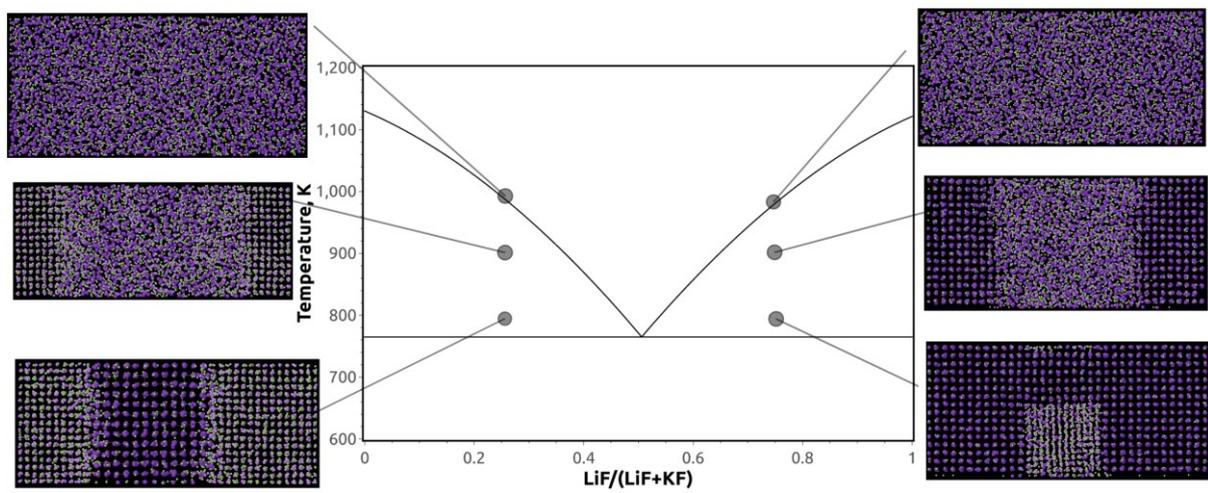


FIG. 90. AMoDy modelling of the LiF-KF system phase equilibrium behaviour.

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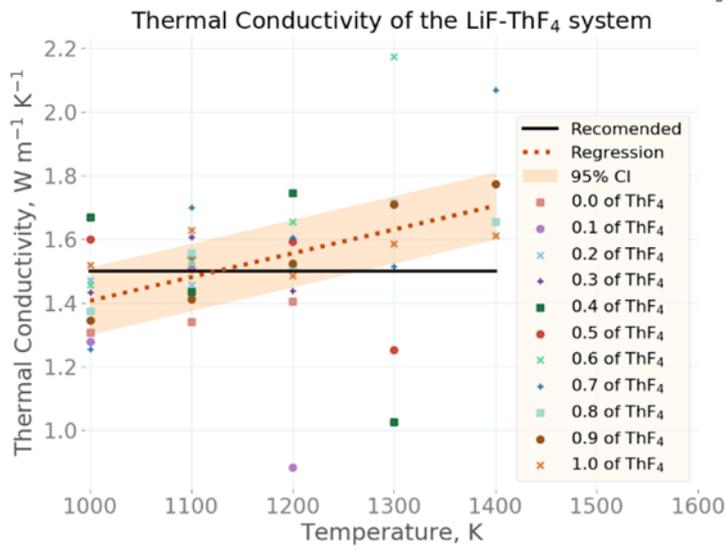
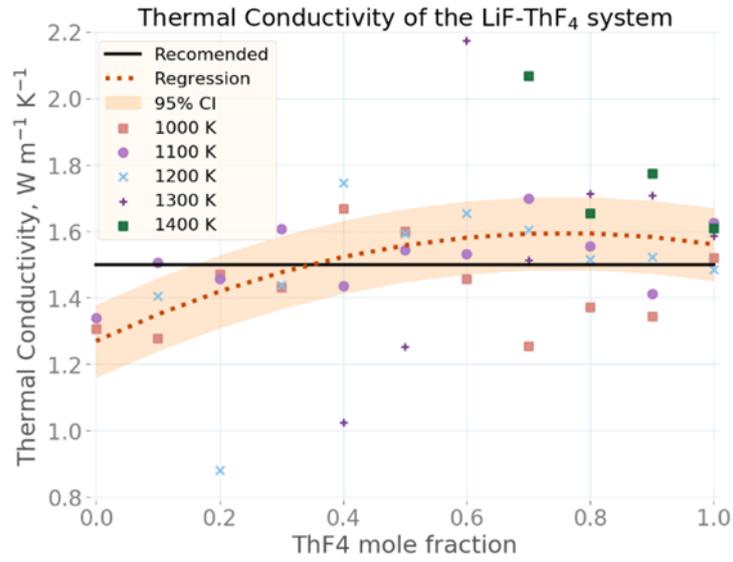


FIG. 91. Thermal conductivity of LiF-ThF₄ system as a function of composition (top) and temperature (bottom). Values calculated by AMoDy and recommended in Ref. [375] are compared.

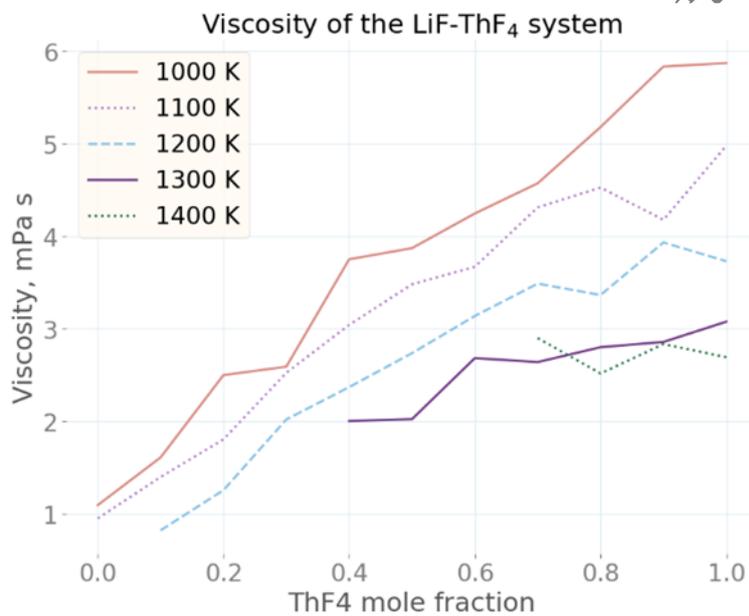
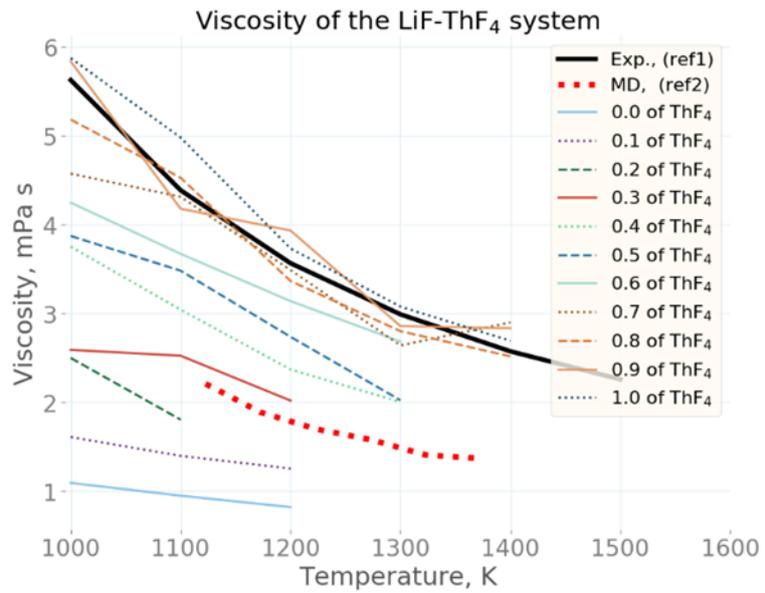


FIG. 92. Calculated viscosity of LiF-ThF₄ system as a function of temperature (top) and composition (bottom). The temperature dependency calculated by AMoDy is compared with values from [375], labelled as (ref1), and [376], labelled as (ref2).

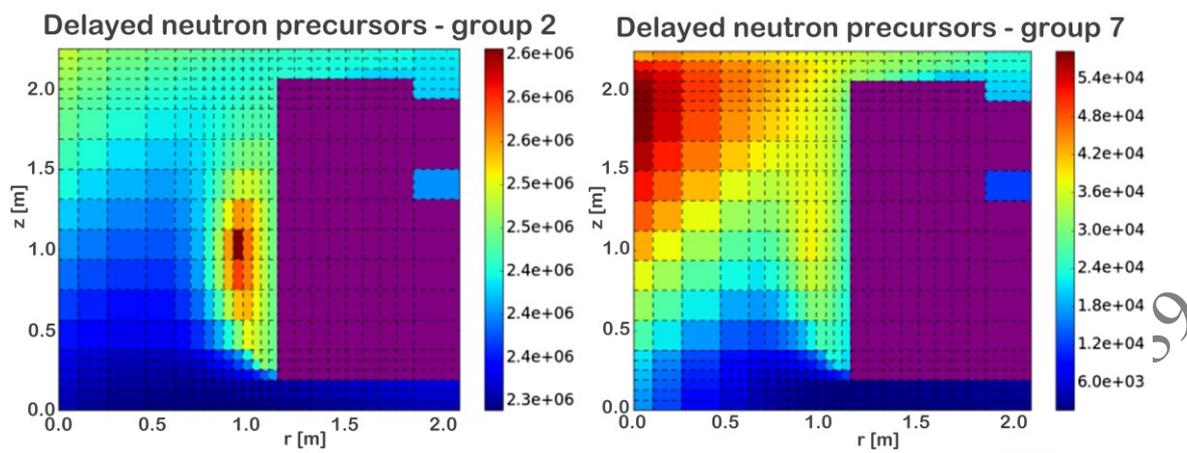


FIG. 93. Distribution of groups 2 (left) and 7 (right) of DNP's in the axial cut of MSFR core benchmark geometry [383].

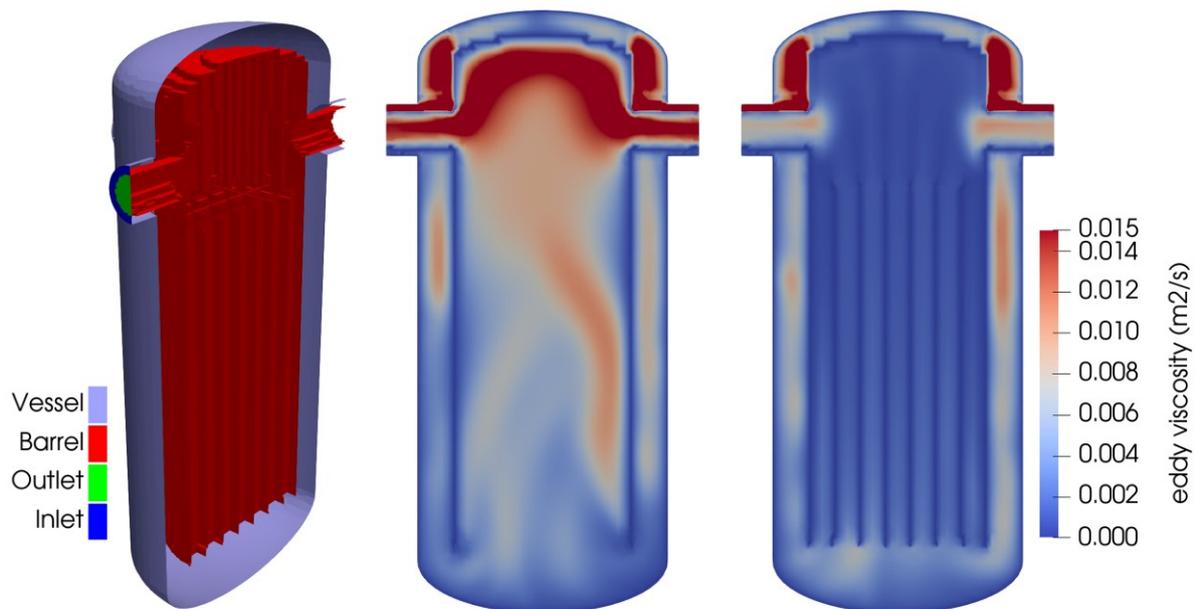


FIG. 94. Impact of baffles on Eddy viscosity in MCFR core. Diagram of baffles (left), Eddy viscosity without baffles (center), and Eddy viscosity with baffles (right) [130].

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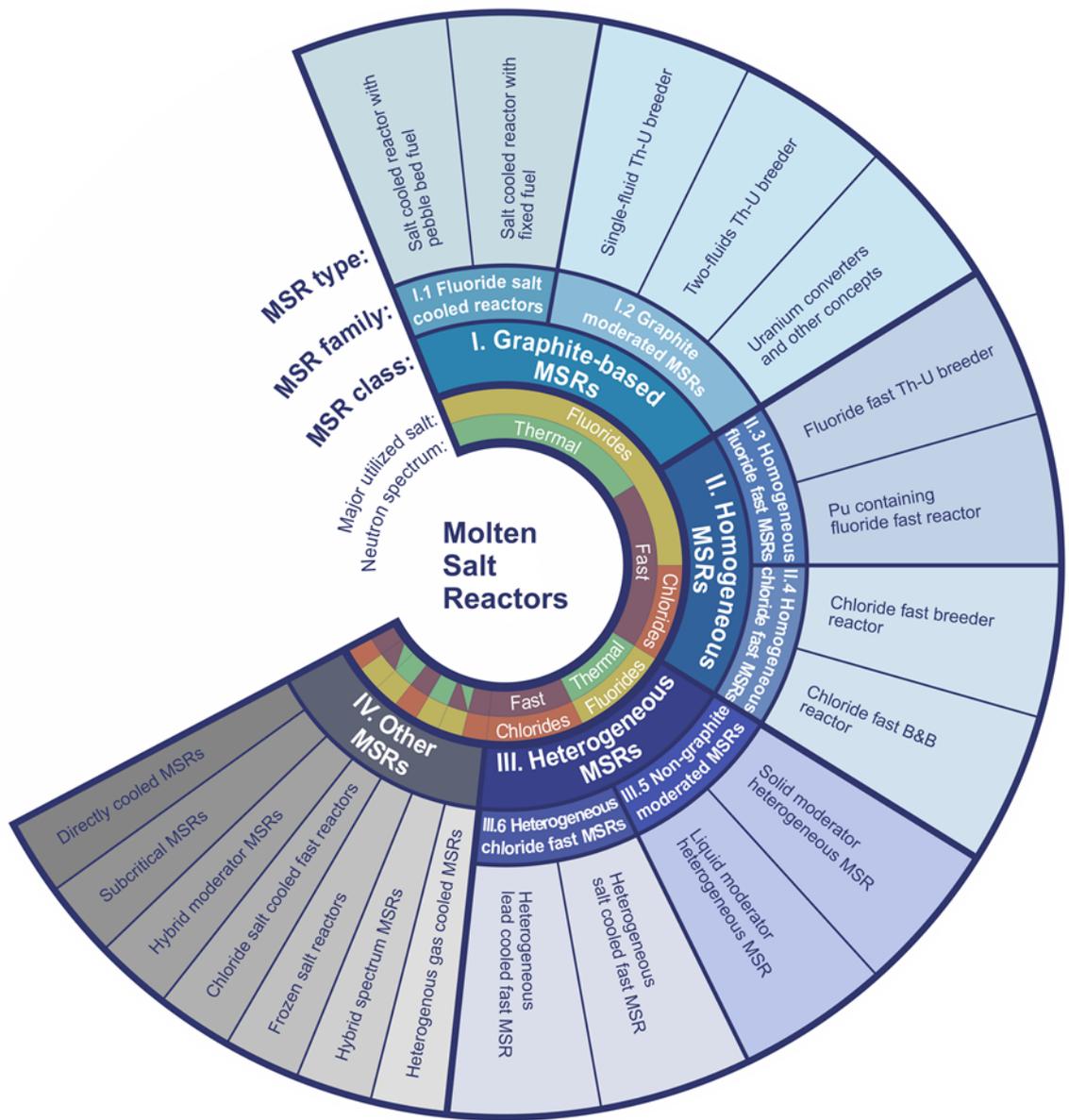


FIG. 95. First three layers of MSR taxonomy (circle view).

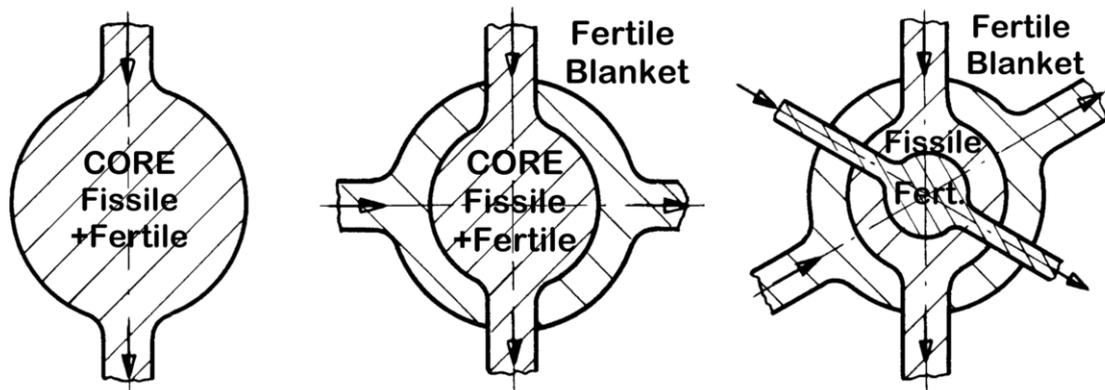


FIG. 96. Basic layouts for homogeneous fast reactors for single-fluid (left), two-fluids (middle) and three-fluids (right) and placement of fissile and fertile materials (Ref. [11]).

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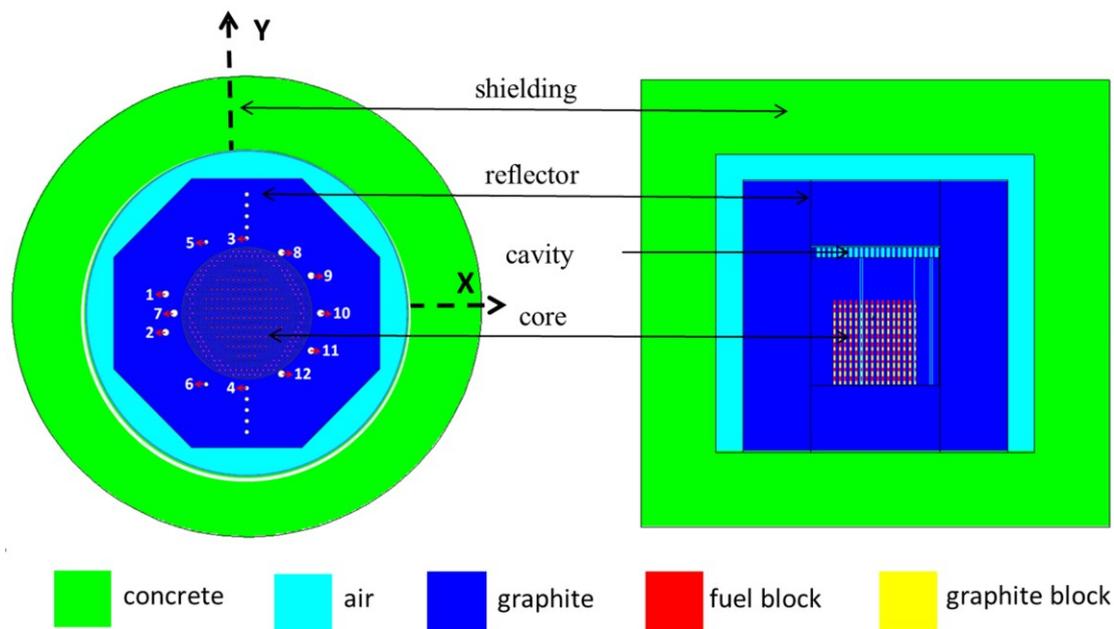


FIG. 97. Schematic diagrams of the Critical Experiment Device. Left and right: Horizontal and vertical cross-sections of the device, respectively. Control rod channels are marked with 1 to 6. Experimental detector and test sample location are marked with 7 to 12.

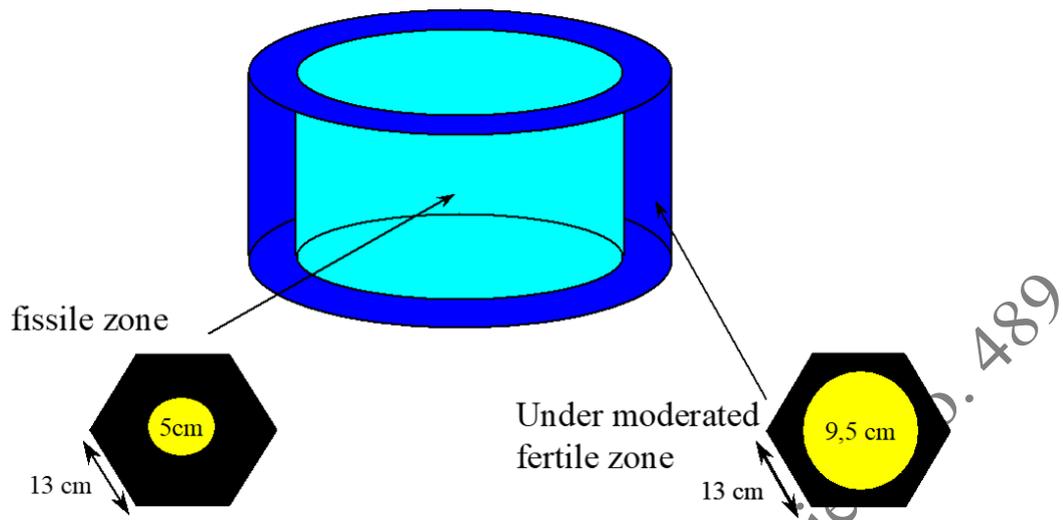


FIG. 98. Representation of the two moderation zones of the AMSTER iso-breeder (black: graphite, yellow: salt).

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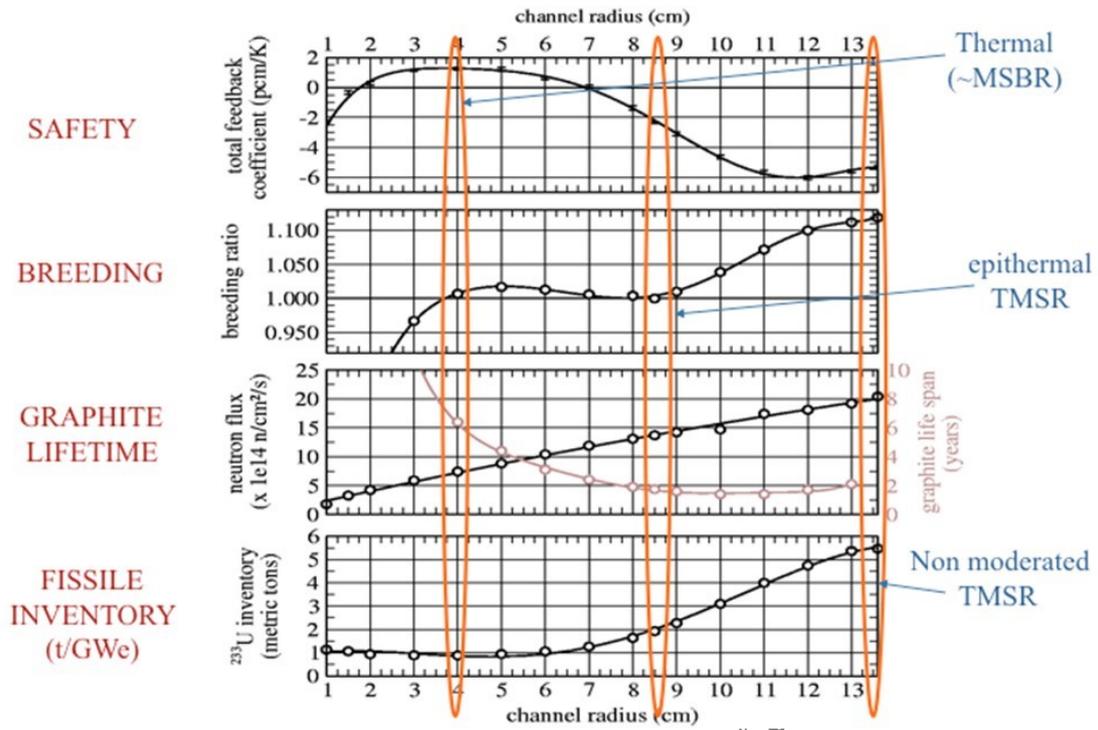


FIG. 99. Main performance parameters for TMSR [59].

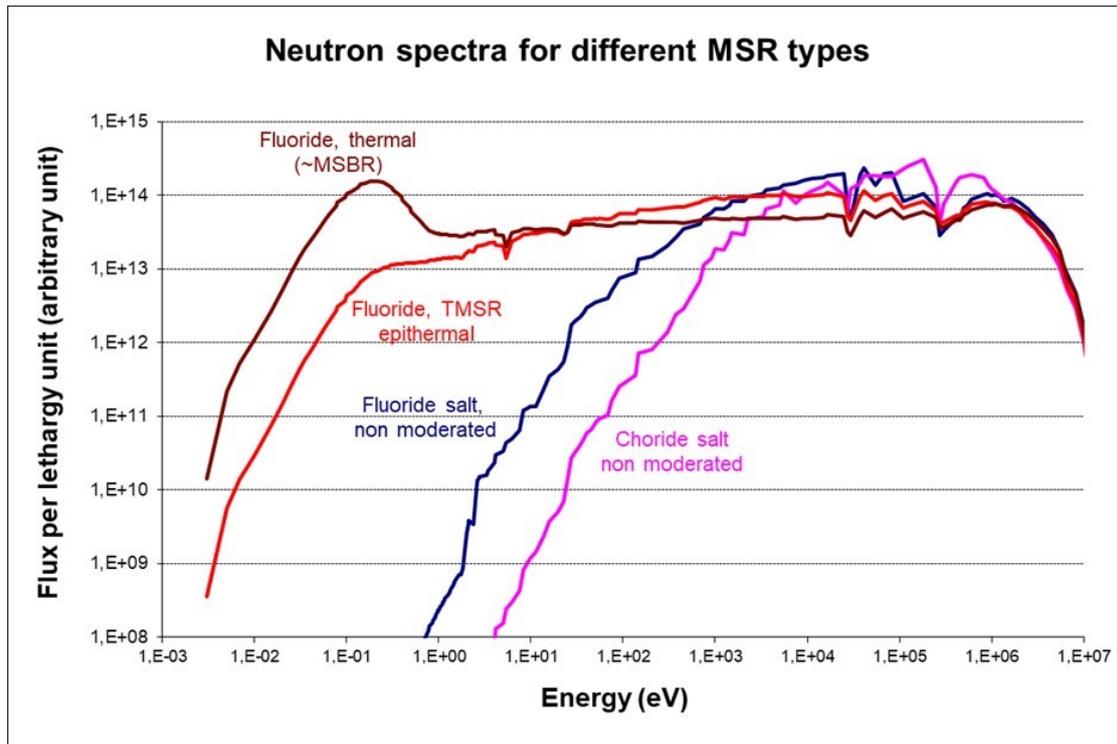


FIG. 100. Neutron spectrum in different MSRs.

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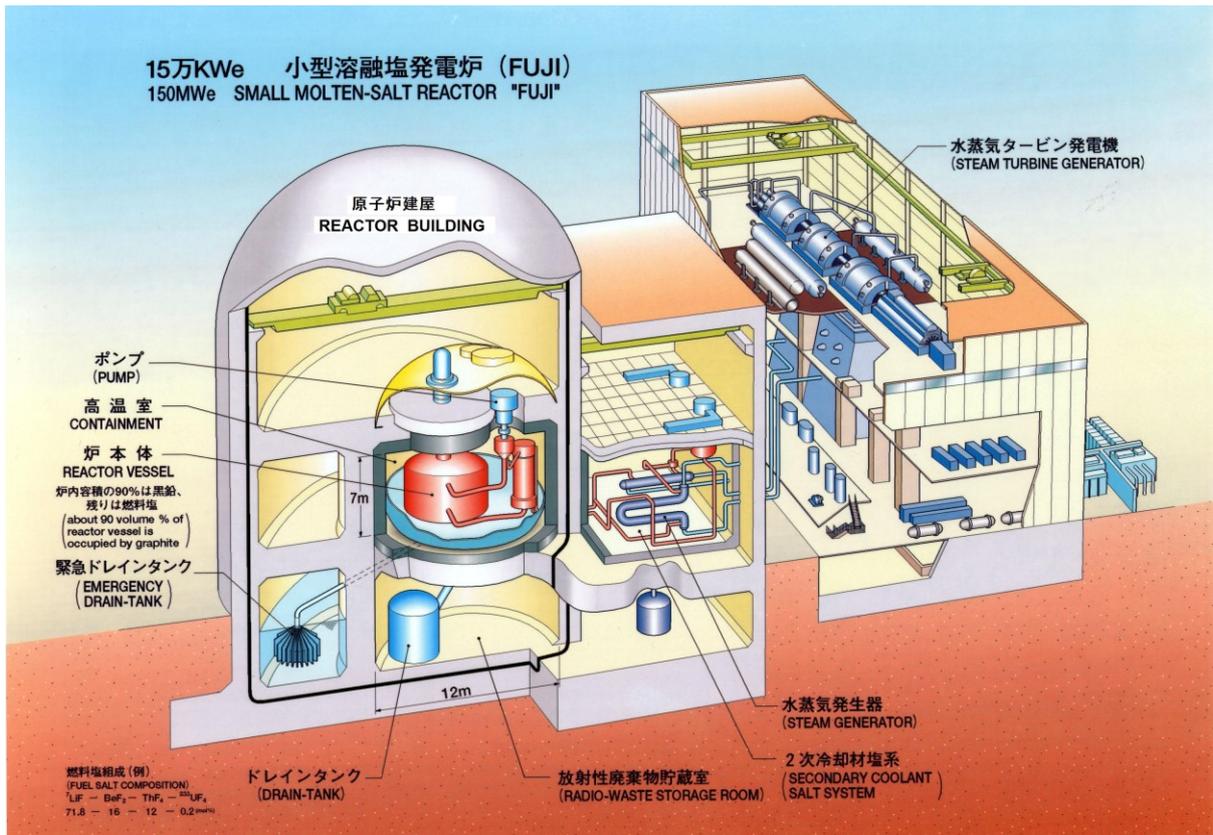


FIG. 102. Aerial view of MSR-FUJI.

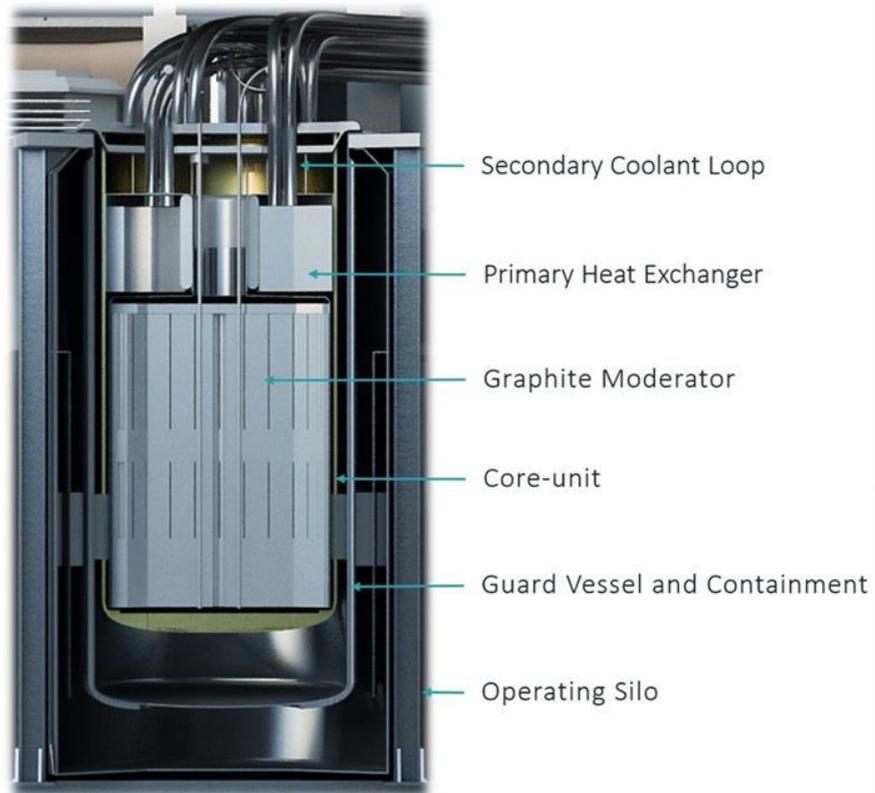


FIG. 103. IMSR Core-unit and its major components shown in an operating silo.

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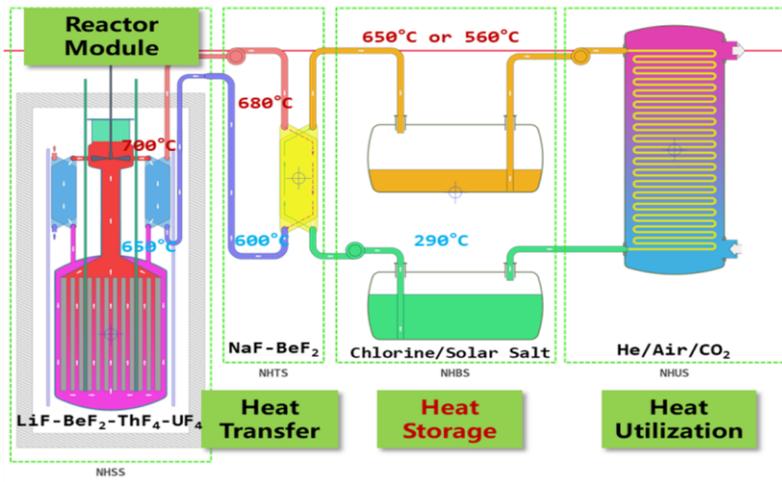


FIG. 104. Diagram of smTMSR-400.

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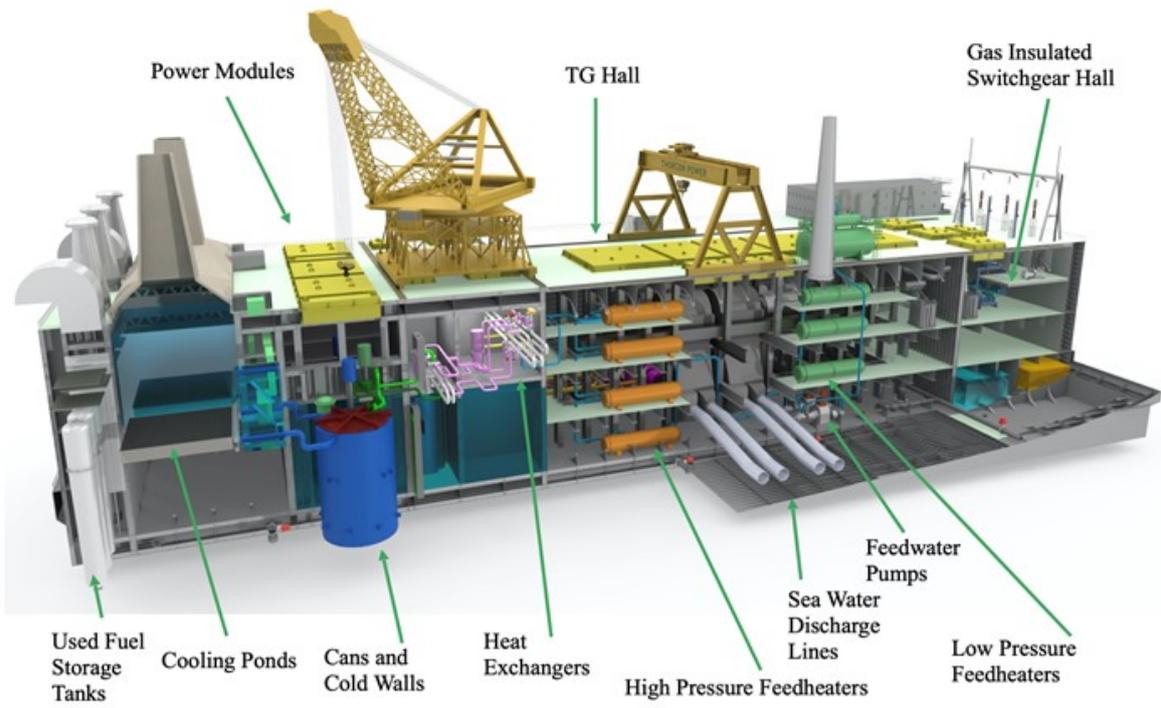


FIG. 105. ThorCon plant.

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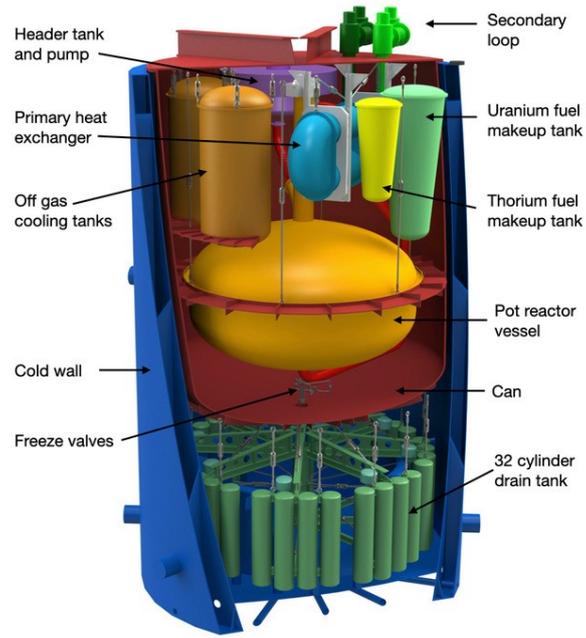


FIG. 106. ThorCon's Pot in Can in Cold Wall.

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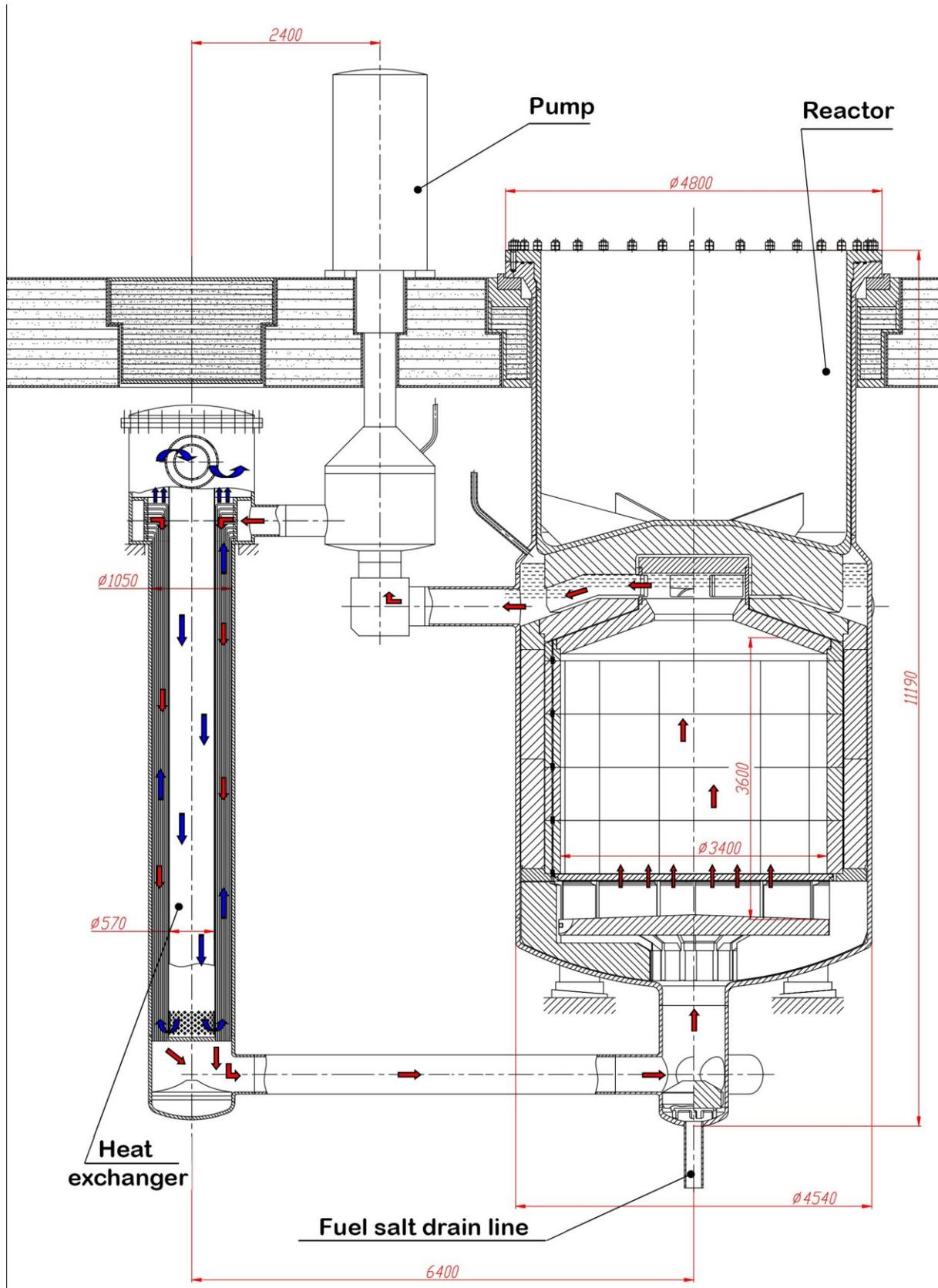


FIG. 107. MOSART fuel circuit.

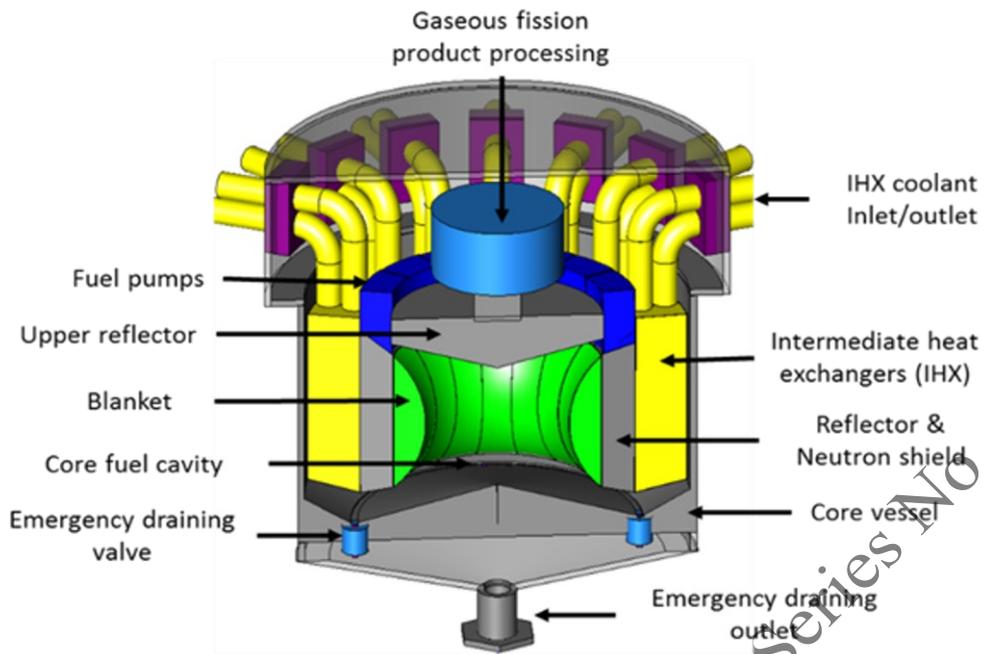


FIG. 108. Schematic layout of the MSFR fuel circuit.



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FIG. 109. Concept illustration of the CMSR primary circuit and fuel salt drain tank.



FIG. 110. Copenhagen Atomics Waste Burner.

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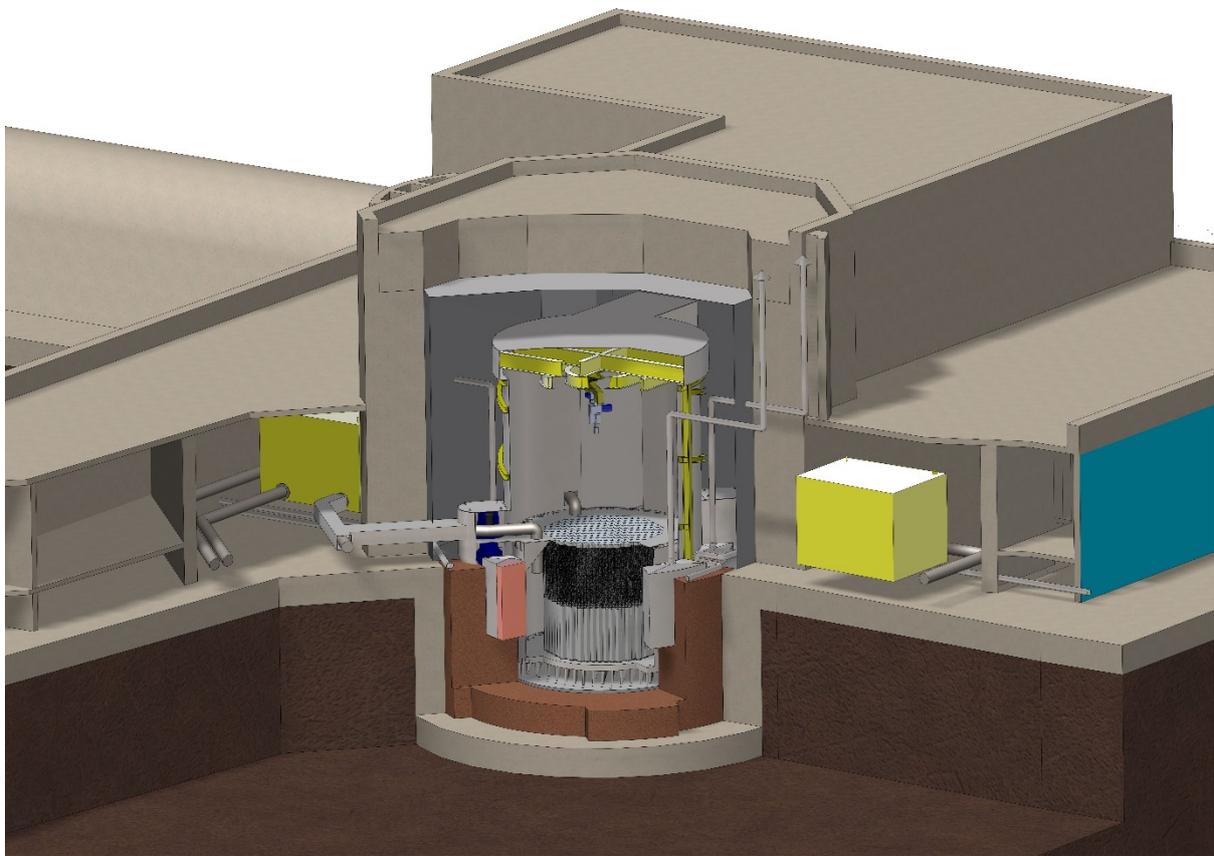


FIG. 111. A sectional view of the SSR-W300.

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