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# Review of Euratom projects on design, safety assessment, R&D and licensing for ESNII/Gen-IV reactor systems

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**Abstract.** Five Euratom projects launched since 2017 in support of the development of ESNII/Generation-IV reactor systems are briefly presented in the paper in terms of key objectives, results, and recommendations for the future. These projects focus on various aspects of the following ESNII/Generation-IV systems: Sodium Fast Reactor, Gas Cooled Fast Reactor, Supercritical Water Cooled Reactor, and Molten Salt Fast Reactor. The paper does not consider EU projects focused on the Gen-IV reactor technologies based on the use of heavy metals as a coolant because these projects are reviewed in a different paper.

## 1 Introduction

The European Sustainable Nuclear Industrial Initiative (ESNII) aims at demonstrating Generation IV Fast Neutron Reactor technologies and supports relevant research infrastructures, fuel facilities, and research and development (R&D) work. ESNII was established in 2010 under the umbrella of the Sustainable Nuclear Energy Technology Platform (SNETP) to promote the European Union's (EU) contribution to Generation-IV reactors.

ESNII supports three technologies and related projects: Lead-cooled Fast Reactor (LFR) and the ALFRED project; Gas-cooled Fast Reactor (GFR) and the ALLEGRO project; Sodium-cooled Fast Reactor (SFR) [1].

The EU framework programs have supported a number of R&D activities on these systems as well as on other Generation-IV technologies, including a European Sodium Fast Reactor (ESFR) and The Molten Salt Fast Reactor (MSFR) as well as some cross-cutting activities. The EU projects focused on LFRs are reviewed in a different paper.

The paper briefly presents in terms of key objectives, results, and recommendations five Euratom projects started since late 2017 in support of the infrastructure and R&D of the four Generation-IV reactor systems (Fig. 1). Table 1 presents the list of the project acronyms, partici-

pants and coordinators. Figure 2 presents the budgets and time spans of the presented projects.

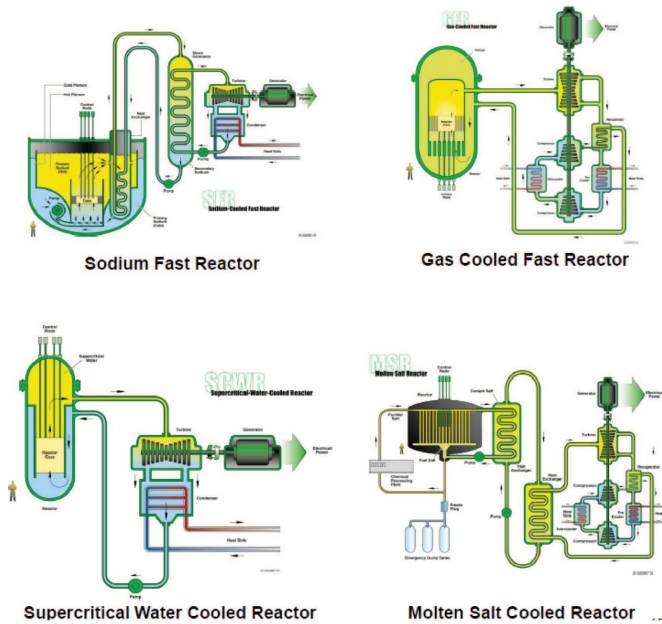
## 2 ESFR-SMART: European Sodium Fast Reactor Safety Measures Assessment and Research Tools

### 2.1 Key objectives

To improve the public acceptance of the future nuclear power in Europe we have to demonstrate that the new reactors have significantly higher safety levels compared to traditional reactors. The ESFR-SMART project [2] aims at enhancing further the safety of Generation-IV SFRs and in particular of the commercial-size European Sodium Fast Reactor (ESFR). The project aims at 5 specific objectives:

- (1) produce new experimental data in order to support calibration and validation of the computational tools for each defence-in-depth level.
- (2) Test and qualify new instrumentations in order to support their utilization in the reactor protection system.
- (3) Perform further calibration and validation of the computational tools for each defence defense-in-depth level in order to support safety assessments of Generation-IV SFRs, using the data produced in the project as well as selected legacy data.

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**Fig. 1.** Four Generation-IV systems supported by the considered EU projects.

- (4) Select, implement and assess new safety measures for the commercial-size ESFR, using the GIF methodologies, the FP7 CP-ESFR project legacy, the calibrated and validated codes, and being in accordance with the update of the European and international safety frameworks taking into account the Fukushima accident.
- (5) Strengthen and link together new networks, in particular, the network of the European sodium facilities and the network of the European students working on the SFR technology.

By addressing the industry, policymakers, and general public, the project is expected to make a meaningful impact on economics, EU policy, and society.

## 2.2 Key results

Originally scheduled for four years the project has been prolonged by one more year in order to accommodate the time delays mainly related to the COVID-19 impact on the experimental activities.

### 2.2.1 Experimental programs

Two specific objectives of the project address new experiments: (1) to produce new data to support calibration and validation of the computational tools for each defence-in-depth level; (2) to test and qualify new instrumentations in order to support their utilization in the reactor protection system. In particular:

- new tests on chugging boiling regime (CHUG) have been conducted to support the computational activities on analysis of the ESFR behaviour under sodium boiling conditions [3].

- New tests on corium jet impingement (HANSoLO) have been conducted using a water-ice system as a model of the corium-catcher system [4].
- The first-ever test (JIMEC-I) on the ablation of a thick steel substrate with a high-temperature, high-velocity steel jet test was successfully performed [4]. With a jet diameter of 40 mm and jet temperature of about 2100 °C, the 40 cm substrate was ablated through the bottom in 31 s. The pooled effect was clearly detected during the course of the ablation.
- Eddy-current flow meters (ECFM) were qualified for positioning above the fuel assemblies in order to detect possible blockages of the sodium flow [5].

### 2.2.2 Benchmarking of codes

One of the specific objectives of the project is to perform further calibration and validation of the computational tools for each defence-in-depth level. In particular:

- a new calculational benchmark has been created for the startup core of the Superphénix (SPX) Sodium Fast Reactor based on open publications [6,7].
- A computational exercise on sodium boiling modelling was conducted based on a KNS-37 sodium loop experiment featuring sodium boiling in pin-bundle geometries [8].

### 2.2.3 Proposal of new safety measures

The key idea is to make the next step in developing the large-power (1500 MWe/3600 MWt) SFR concept, following up the “line” of the Superphénix 2 (SPX2), European Fast Reactor (EFR) and ESFR designs and using the set of the GIF objectives as a target. In particular:

- the ESFR core design modifications were aimed at improving the core map symmetry; optimizing the void effect, and facilitating the corium relocation toward the corium catcher [9].
- The ESFR system modifications were aimed at simplifying the overall design (see Fig. 3) and improving the safety functions: control of reactivity, heat removal from fuel, and confinement of the radioactive materials [10].

### 2.2.4 Evaluation of core performance

After the new core design was proposed studies were launched to check how this core design will influence the neutronics and fuel performance. In particular:

- six-batch burnup calculations were performed using a Monte Carlo code and the core state specification at the End of the Equilibrium Cycle was defined, including the 3D isotopic composition needed to calculate the reactivity coefficients and kinetics parameters as well as the 3D power distribution for the following-up thermal-hydraulic analysis [11].
- Fuel performance for a typical cycle was analysed with a number of fuel performance codes and the gap heat conductance correlation was derived for the subsequent steady-state and transient thermal-hydraulic analyses.

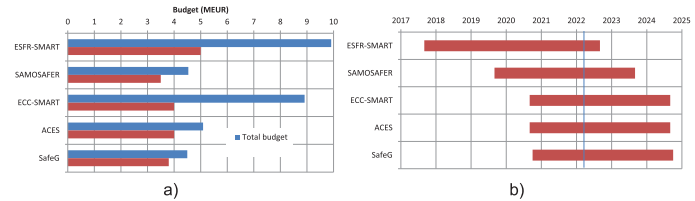
**Table 1.** Participants and coordinators of the considered EU projects (C = Coordinator, X = Partner).

	BE	CH	CZ	DE	ES	EU	FI	FR	HU	IT	LV	NL	PL	RO	SE	SK	SI	UA	UK	SUM			
	SCK-CEN ENGIE LABORELEC	PSI UIV REZ EVALION	VSCT CTU CV REZ	GRS KIT HZDR	BGF UPM CIEMAT	JRC ENEN	VTT IRSN	CEA FRAMATOME	EDF LGI	UNI DE LORRAINE EK-CER	BME ENEA	POLITO POLIMI UNI PISA	IPUL NRG	TU DELFT NCBJ	RATEN	KTH CHALMERS	VUJE STU	IJS ZAG	IPP CENTRE LLC ENERGORISK	JACOBS UNI CAMBRIDGE UNI NOTTINGHAM UNI SHEFFILD	NNL	SUM	
ESFR-SMART		C		X	X	X		X	X	X	X	X	X		X				X	X	X	19	
SAMOSAFER		X		X		X		X	X	X	X									X	X	X	12
ECC-SMART			X	X		X	X	X			X			X	X		X	X	X		X	X	16
ACES	X	X		X	X			C	X	X								X					10
SafeG			X	X	X	X			X					X			C	X		X	X	X	14

### 2.3 Recommendations for the future

A list of recommendations for the research and development of ESFR was prepared and discussed and on this basis, a new European project ESFR-SIMPLE was submitted and approved by European Commission. The new project is planned to start at the end of 2022, providing therefore a continuity of research and development. ESFR-SIMPLE stands for “European Sodium Fast Reactor – Safety by Innovative Monitoring, Power Level flexibility and Experimental research” and aims at challenging the current ESFR design to improve its safety and economics through the implementation of innovative technologies in accordance with the ESNII roadmap. A small-power version of ESFR is planned to be developed in this project. A special focus of ESFR-SIMPLE on further improvement of safety and economics was set based on the conclusions derived in ESFR-SMART (in particular, such conceptual design simplifications as replacement of the safety vessel by the liner on the pit surface, integration of the decay heat removal systems in the secondary circuit and more compact design of the secondary system will be studied for the low-power option). The project has 5 specific objectives:

- (1) rethink the ESFR design in order to simplify it and make it more cost-competitive, while still achieving resource sustainability and having safety reinforced by intrinsic behaviour. This can be accomplished by reducing the size of the reactor, which will also allow taking advantage of small modular reactor features such as transportability, modularisation, standardisation, and flexible operation, all ultimately leading to improved economics.
- (2) Assess the impact of alternative technologies, such as metallic fuel and compact secondary system design, for the large-size ESFR on the economics and safety.
- (3) Propose, develop and assess advanced methods of monitoring and processing operational data using Artificial Intelligence, e.g., to optimise fault detection in steam generators at an early stage.
- (4) Produce new experimental data in order to assist in the qualification of innovative components, such as expansion bellows, core catchers and thermo-electric pumps.



**Fig. 2.** Budget (a) and time span (b) of the considered EU project.

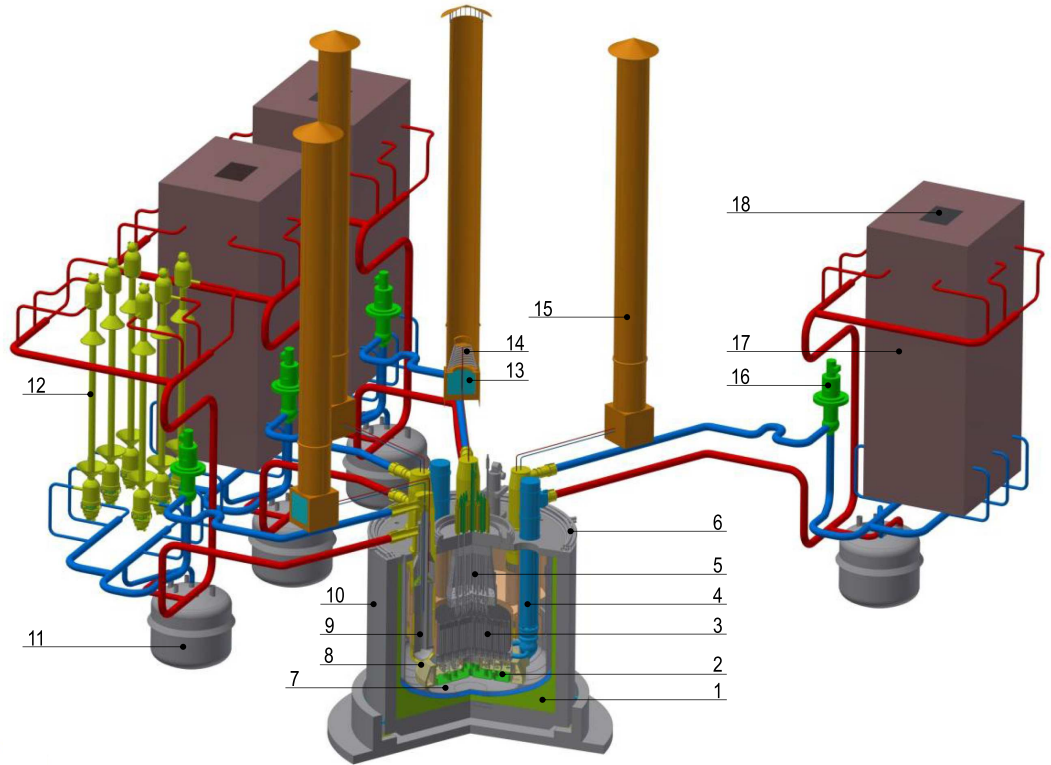
- (5) Ensure that the knowledge generated in the project is shared not only among the project partner institutions but also with a wide range of stakeholders in Europe and internationally. The project activities will also be informed by the public and other stakeholders’ perceptions of the risks and benefits of ESFR technology.

## 3 SAMOSAFER: Severe Accident Modeling and Safety Assessment for Fluid-fuel Energy Reactors

The ultimate aim of nuclear energy research is to develop a nuclear reactor that is inherently safe and that produces no nuclear waste other than fission products. The Molten Salt Fast Reactor (MSFR) has the potential to reach these goals. The most characteristic property of the EU MSFR design is the liquid fuel, which provides excellent options for reactivity feedback and decay heat removal. Furthermore, the continuous recycling of the fuel salt enables one to design a reactor either as a breeder reactor with in-situ recycling of all actinides or as a burner capable of incinerating the actinide waste from other reactor types. For these reasons, the SNETP has emphasized in its strategic research agenda of 2012 (Annexes on Thorium and Molten Salt Reactors) the merits of the MSR as the long-term option for safe and sustainable nuclear power.

### 3.1 Key objectives

The goal of SAMOSAFER is to develop and demonstrate new passive safety barriers for more controlled behaviour



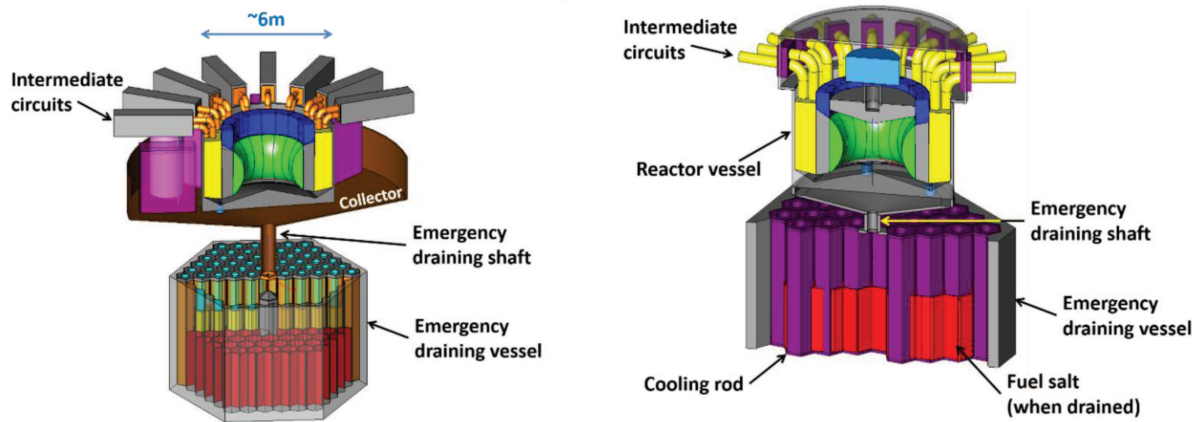
- |                                |                           |   |
|--------------------------------|---------------------------|---|
| 1: Insulation with steel liner | 7: Main vessel            | 13: Window for air circulation (DHRS-1) |
| 2: Core catcher                | 8: Strongback             | 14: Sodium-air HX (DHRS-1)              |
| 3: Core                        | 9: IHX                    | 15: Air chimney (DHRS-1)                |
| 4: Primary pump                | 10: Reactor pit           | 16: Secondary pump                      |
| 5: Above-core structure        | 11: Secondary sodium tank | 17: Casing of SGs (DHRS-2)              |
| 6: Pit cooling system (DHRS-3) | 12: Steam generator       | 18: Window for air circulation (DHRS-2) |

**Fig. 3.** General view of ESRF-SMART reactor.

of MSR in severe accidents, based on new simulation models and tools validated with experiments. The grand objective of SAMOSAFER is to ensure that the MSR can comply with all regulations expected in 30 years' time.

After the successful completion of the project, the simulation models and tools can be used by the nuclear industry, and the innovative safety barriers can be implemented in new MSR designs like the MSFR (Fig. 4). This will lead to increased safety margins in future Gen-IV reactors to ensure they will comply with future more stringent safety standards. The grand objective has been divided into the following specific sub-objectives:

- (1) investigating and translating the existing defence-in-depth safety approach to MSR.
- (2) Developing a rigorous and well-established simulation code suite through:
  - (a) developing theoretical models of physics and thermo-chemical phenomena relevant to the safety assessment and severe accident analysis in MSR, such as salt solidification and melting models;
  - (b) developing simulation models and tools to be included in cutting-edge computation codes;
  - (c) coupling existing computation codes to deliver an integrated simulation approach to complex phenomena, such as neutronics-CFD coupled to thermo-chemistry;
- (d) simulation models for the computation of multi-physics phenomena in existing codes.
- (3) Developing and applying experimental setups for the validation of simulation models and tools through:
  - (a) developing new experimental facilities for the validation of new theories and new models as well as for the validation of existing codes to new applications;
  - (b) modifying and applying existing experimental facilities to generate data for validation purposes;
  - (c) using experimental setups and computational schemes to generate data for the extension of user databases, such as the molten salt database at JRC-Karlsruhe.
- (4) Design of advanced barriers for severe accidents in MSR, such as freeze plugs, drain tanks, and fission product extraction/immobilization methods, as well as the demonstration of the effectiveness of these barriers.
- (5) Update of the MSFR design with all improvements and recommendations from aforementioned studies.



**Fig. 4.** Schematic design of the primary circuit of the MSFR showing the reactor vessel and Emergency Draining System for the fuel salt.

- (6) Attracting and educating students, postdoctoral researchers and trainees in the exciting field of MSR and advanced Gen-IV nuclear energy.
- (7) Develop and train a software user community and deliver the results of our project to future users to speed up the development of the MSR value chain.

Besides the Work Package (WP) on project management (WP8), the SAMOSAFER project contains seven specialized parts.

WP1 lays the foundation for the safety approaches applied in WPs 2–6. It compares the EU nuclear safety standards with the MSR safety case and extrapolates these to future requirements, identifies the risks, and the postulated initiating events for the Fuel Treatment Unit (FTU) like was done for the core and the Emergency Draining System (EDS) in the SAMOFAR project. This WP also explores and gives guidelines for the safety approaches to be taken for the nuclide retention in WP2, the source term redistribution in WP3, the radionuclide confinement in WP4, the decay heat removal in WP5, and the reactivity control in WP6. Furthermore, a global overview of integral experiments will be made to provide benchmark cases and to identify the needs for integral experiments. This WP is carried out with the strong involvement of the nuclear industry and TSO.

The aim of WP2 is to provide tools and data necessary to control the fuel salt behavior in the MSR and to assess the influence of fission products and corrosion products on the fuel salt properties. To this end, we extend our current code systems (DG-FLOW and OpenFoam) developed in SAMOFAR with a new thermochemistry module and perform extensive experimental validation and modelling. We perform Post Irradiation Examination (PIE) on the SALIENT-01 samples irradiated in the High Flux Reactor (HFR) in the Netherlands to study fuel salt behavior (fission product production, speciation, and relocation), the particle size distribution of noble metals, etc. of the irradiated fuel salt. Other properties will be measured as well, complemented with molecular dynamics simulations.

In WP3 we evaluate the nuclide inventory through the whole reactor including the FTU. The equilibrium

nuclide concentrations in the core depend on the treatment scheme, therefore a new coupled code system simulating fuel burnup and chemical extraction methods will be developed for a complete and accurate safety assessment. Simulation tools will be used to assess the extraction efficiencies of the gaseous fission products and micro-particles (noble metals) via helium bubbling. The chemical extraction processes in the FTU will be evaluated experimentally and assessed with the new simulation codes.

Essential for resistance against severe accidents is the ability of fuel salt to flow freely to redistribute the decay heat and to bring the reactor to a safe condition under abnormal operating conditions. Because heat conduction in the fuel salt is generally poor compared to heat convection (high Prandtl number), the fuel salt may easily solidify against cold walls or in cold regions. In WP4 we develop cutting-edge simulation tools to evaluate the melting and solidification phenomena of the fuel salt. We use these to optimize the Emergency Draining System (EDS) and other barriers including the freeze plugs and valves. The software will be validated using the experimental facilities SWATH-S and ESPRESSO.

In WP5 we develop simulation models and tools for the safe removal of decay heat from the fuel salt in the reactor core and the EDS. Because the fuel salt is partially transparent to infrared radiation, radiation heat transfer is more important for decay heat removal than in solid fuel reactors. On the other hand, natural convection is less effective, due to the internal heat production in the fuel salt, which leads to smaller density differences. We study both the natural convection phenomena, the effects of turbulence on heat transfer, and the radiation heat transfer mechanisms including experimental validation using SWATH-S and DYNASTY. The resulting models will be used to optimize the fuel circuit and the EDS with regard to decay heat removal.

In WP6 we integrate the results of our project by applying our simulation models and tools to new safety barriers. First, we define the MSFR operational states as well as the emergency operating procedures. Subsequently, we investigate advanced monitoring and predictive control strategies to prevent severe accidents. By control of

the redox potential of the fuel salt, e.g., by controlling the ratio of  $\text{UF}_4$  and  $\text{UF}_3$ , corrosion can be reduced, increasing the long-term mechanical strength of the fuel circuit. The barrier designs from WP3–5 will be incorporated to show their effectiveness with regard to risk reduction and severe accident prevention including uncertainty quantification. The effects of scaling laws on the occurrence and effects of severe accidents will be investigated to find generic relations for the design of MSR. This knowledge will be transferable to other reactor designs as well.

WP7 focuses on the education and training of doctoral students and postdoctoral researchers in the project, and on the activities to maximize dissemination and exploitation. We organize a summer school and a Young-MSR conference for students, trainees, postdocs, scientists and engineers, and address an even wider community via webinars and other means such as a software simulator. To widen the skillset of our students, we support and stimulate the exchange of students and trainees. To build a software user community for our simulation tools, we will organize an exploitation workshop with hands-on lessons using the software. The exploitation of our results will be targeted at strategic stakeholders needed for the further development of the MSR.

### 3.2 Key results

In WP1 we defined the notion of severe accidents in Molten Salt Reactors (MSR) based on the methodologies from the Risk and Safety Working Group of GIF leading to so-called severe plant conditions. Based on this evaluation a specific MSR-oriented defence in-depth approach has been set up by analysing the safety functions of all fuel salt locations in the reactor and by defining the number of containment barriers.

A list of MSR-specific Postulated Initiating Events (PIE's) with a focus on reactivity insertion events have been established and a simulation tool for the reactor has been developed treating slow and fast dynamics. To establish the list of PIE's for the Fuel Treatment Unit (FTU) the process diagram has been reorganized to provide input to the Functional Failure and Mode Effect Analysis (FFMEA) methodology, which will be executed in the second half of the project.

The review of (integral) experiments for the validation of MSR code systems has started as well as an analysis to arrive at Process Identification and Ranking Tables (PIRT) to prioritize various phenomena important for the validation and demonstration of MSR.

In WP2, the Thermochemica software has been coupled to the JRC-Molten Salt Data Base for thermodynamic assessments of various salts. At a later stage, Thermochemica will be coupled to multi-physics codes as well. In the fuel salt, noble metals appear as solid precipitates. A single-phase Eulerian approach has been applied to model the transport and extraction of these particles. For better results, more accurate estimates of the noble metal particle size distribution are needed.

A thermodynamic assessment of various salt compositions has been performed by the Calphad method.

Validation was performed by experimental studies on  $\text{CrF}_3$ . After irradiation of the  $\text{LiF-ThF}_4$  samples in the High Flux Reactor (NL), the post-irradiation experiments started. The graphite crucibles have been punctured and the plenum gas evaluated. Further evaluation of the irradiated samples will be done in hot cells at NRG and JRC-Karlsruhe.

Furthermore, molecular dynamics studies were done on  $\text{LiF-ThF}_4$  using a new forced-field model. They turned out to be capable of reproducing the experimentally observed viscosity. Other studies use the JRC database to predict properties like the density and viscosity of multi-component fuel salts. A computational tool has been developed to interpolate and extrapolate density and viscosity data, which can be used in thermal hydraulics codes.

Preliminary results of the benchmark calculations show that a substantial part of the decay heat can be generated in the off-gas system. The benchmark calculations simulating the fission product inventory with various processing schemes show good agreement. Multiphysics solvers have been extended for the simulation of gaseous fission products. Experimental studies on metallic particle extraction via bubble flotation are still ongoing.

Thermodynamical calculations resulted in a preliminary scheme for reprocessing chloride salts, and literature studies on the immobilization of fluoride and chloride salts were done. Several issues in the treatment of chloride salts by electrolysis were identified. The assessment of the fused salts volatilization method was limited to corrosion tests on selected nickel alloys and the fabrication of some apparatus. Simulations have been executed to evaluate fission product releases from the fuel salt under accident conditions.

Modelling freezing and re-melting of the salt is challenging because of the displacement of the solid-liquid interface. Several enthalpy approaches have been tested in multiple codes. Benchmarking of convection models with literature was successful, but the further implementation of phase change models needs to be done. Experimental data on phase change phenomena in a forced or mixed convection flow regime are not as abundantly available as data on phase change phenomena in conduction or in natural convection mode. In the near future, we expect to generate new experimental data for validation.

The SIMMER code has been extended and input has been prepared for the simulation of the Castillejos experiment for validation. Also, the secondary circuit and the heat exchanger have been included in the SIMMER model. Calculations with SYRTHES on conductive heat transfer showed good agreement with the literature. Studies showed that residual heat extraction by airflow only might not be sufficient and that other means should be sought.

The SWATH-S experimental facility has been extended to study radiation heat phenomena in salt. Two extensions, one closed and one open channel, have been designed and the closed channel has been built and installed. The experimental campaign has just started. Numerical models to study radiative heat transfer have been developed and will be validated when experimental data becomes available. DYNASTY will soon be used to

measure natural convection and heat transfer phenomena which will be used to validate numerical models. The natural convection stability analyses performed in SAMOFAR will be extended in SAMOSAFER to provide more insight into the stability analysis of MSR.

In WP6, design drawings of the core and passive decay heat removal system and enclosing building structures have been made to get a clear view of these structures. The normal operating transients (startup/shutdown/etc) of the MSFR and the main plant parameters have been identified. These are inputs to the design of the MSR simulator needed to investigate predictive control strategies. The experimental protocol to analyse all elements at once in the fuel salt composition has been defined, but the experimental work as well as the measurement of the fertile and fissile components in the salt via electrochemical methods have been delayed due to Covid.

The summer school was held online in the second half of 2021 and will be finalized jointly with the Young MSR conference in June 2022. It attracted more than 200 students from over 30 nationalities.

The publications of the SAMOSAFER project published in the open literature till now can be found in references [12–21].

### 3.3 Recommendations for the future

The EU MSFR is a reactor design at a low TR level with many remaining points for improvement. We made significant progress in the design and safety analysis of this reactor using a newly developed MSR-specific safety approach, but for a more detailed design, we need to extend our knowledge of materials data (structural materials, fuel salt properties, corrosion data, etc), extended code packages with improved validation in extreme situations, non-proliferation issues, modularization and maintenance of the design, safety by design, the location and chemical form of the fuel inventory (both in the reactor core and the Fuel Treatment Unit), etc.

## 4 ECC-SMART: Joint European Canadian Chinese Development of Small Modular Reactor Technology

The ECC-SMART project is supported by Horizon 2020 Euratom call and focuses on a small modular reactor cooled by supercritical water (SCW-SMR) with the aim to support and help to solve the issues relating to structural materials, heat transfer, neutron physics and safety. The results will be used for assessing the feasibility and identification of safety features of SCW-SMR, considering specific knowledge gaps related to the future licensing process and implementation of this technology.

The project is divided into four technical work packages (material testing, thermal-hydraulics and safety, neutronics and reactor physics, and guideline synthesis and pre-licensing studies) and two administrative work packages, including nuclear education and training activities.

Altogether, they aim to achieve the main objectives to define the design requirements for the future SCW-SMR technology, developing the pre-licensing study and guidelines for the demonstration of the safety in the further development stages of the SCW-SMR concept including the methodologies and tools to be used and to identify the key obstacles for the future SMR licencing and propose a strategy for this process.

The project consortium has 20 partners and consists of 15 European institutions from 12 member states and is supplemented by an important international collaboration with the Canadian, Chinese, and Ukraine supercritical water research programs. The project consortium and project scope were created according to the joint research activities under the International Atomic Energy Agency, Generation-IV International Forum, SNETP and NUGENIA umbrella. Hence, ECC-SMART maximizes international synergy between the national programs of the European member states on the one hand and the international programs on the other hand.

### 4.1 Key objectives

In general, the objective of the project is to provide science-based recommendations, methodologies for performing safety evaluations and safety improvements fostering the safety standards, including the experimental validation of essential items for safety demonstrations related to the SCW-SMR.

The activities of the project will lead to deriving the most economical safety-driven supercritical water-cooled small modular reactor (SCW-SMR) design requirements by identifying adequate solutions to key technical issues which drive cost and safety and their influence on the future licencing process. An important feature of the project is also the ambition to increase the level of knowledge as well as the interest of the industrial partners and demonstrate the benefits of the SCW-SMR concept.

However, the key objectives represent the main structure of the project:

- (1) complete the understanding of the corrosion behaviour of the most promising candidate materials at different conditions to support the qualification procedure of the future SCW-SMR constructional materials and assess the relation to the existing standards and guidelines.
- (2) Define the design requirements for future SCW-SMR technology. Verify, validate and further develop the selected thermo-hydraulic system-, subchannel-, safety-, and CFD-codes and assessment of the proposed SCW-SMR concepts by applying these codes.
- (3) Provide reactor physics analysis of preliminary core layout based on a design proposed in the project including selection of proper neutron/reactor physics code.
- (4) Develop pre-licensing studies and guidelines for the demonstration of safety in the further development stages of the SCW-SMR concept. Identify the key obstacles to the future SMR licencing and propose a



strategy for this process based on the findings on the multinational level.

In addition, the project wants to progress education and training and help increase interest in the nuclear field among young scientists. For this purpose, events, and workshops on variable topics from the nuclear field are and will be organized under ECC-SMART.

## 4.2 Key results

The project reached the first third of its duration, thus, most activities and corresponding results are expected in the next months. However, the overview of the project's main activities, relating to progress and results is described below.

The biggest work package is focused on material testing (WP2). For this purpose, the test matrix was established including more than 700 specimens. The focus is devoted to stainless steel 310S and alloy 800H, which have been selected as the most perspective material for fuel cladding for SCWR. In addition, experimental material such as AFA (alumina forming austenitic alloy) was supplied by colleagues from China (USTB). Most of the tested specimens have been manufactured from the tubes to get closer to the real conditions and to support the progress in the SCWR and SCW-SMR fields. The first planned experiments have been started and the results are being evaluated.

Within the work package, 3 on thermal hydraulics and safety of the SCW-SMR, an innovative design concept of a small modular reactor cooled by SCW (see Fig. 5) has been proposed based on HPLWR (high-pressure light water reactor) including the experiences from Canadian and Chinese designs. The first corresponding results have been published at the ISSCWR-10 conference [22]. The main design of the core is more complex as 6 heat-up stages of the moderator have been proposed. Accompanied sketches in Figure 5 show some basics of design such as the horizontal placement of fuel assemblies and the different heat-up stages achieved by horizontal plates welded with the tube sheet (Fig. 5b). On top of the core is a steam plenum above the reflector room, from where the superheated steam is leaving via two outlet flanges (Fig. 5a) [22]. Thus, the coolant flows inside the reactor core are always horizontal or upwards. However, further tests focused on thermo-hydraulic data and supporting the innovative design were proposed and started [23,24]. A conceptual investigation of passive safety aspects is prepared and will be performed.

Based on the proposed innovative design by the WP3, the first calculations relating to the reactor physics have been carried out as part of work package 4 (neutron physics of SCW-SMR). A few modifications have been proposed and they are under discussion to achieve a satisfactory design including economical and feasibility aspects. In addition, a computational benchmark was developed and used for the comparison of different neutron codes.

The review report on safety criteria and requirements for the SCW-SMR concept was completed within work

package 5 (synthesis and guidelines for safety standards). A compilation of the SSS (safety, security and safeguards) features implemented in the SMR is already being in advanced design and especially licensing stages will be further developed during the ECC SMART project including the progress performed in relevant work packages 2–4 and would propose a consistent set of the main safety-related findings and conclusions.

## 4.3 Recommendations for the future

The ECC-SMART project is at the beginning and key results are ahead of us. The actual geo-political situation uncovered the lack of appropriate sources of energy supporting its diversification and the independency of countries. In this light the importance of relevant research oriented on the future supplement of energy is necessary. Hopefully, the achievement of expected results would not be significantly affected.

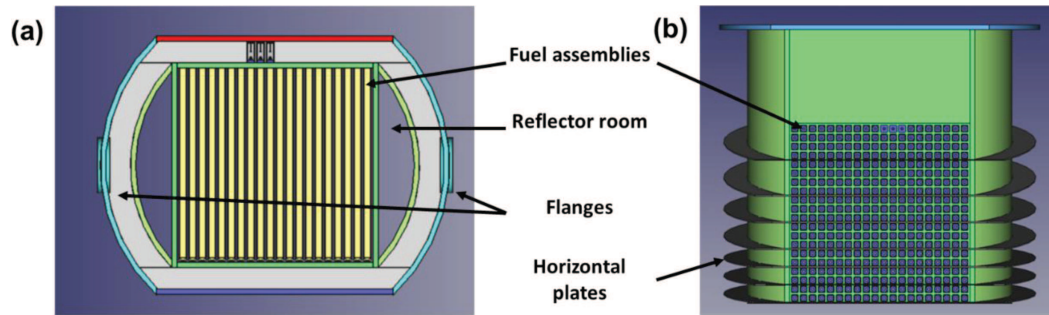
# 5 ACES: Towards Improved Assessment of Safety Performance for Long-Term Operation of Nuclear Civil Engineering Structures

## 5.1 Key objectives

ACES addresses the EURATOM Work Programme 2019–2020, dedicated to Nuclear Fission and Radiation Protection Research regarding nuclear safety namely through ageing phenomena of components and structures and operational issues.

The main objective of ACES is to advance the assessment of the safety performance of civil engineering structures by solving the remaining scientific and technological problems that currently hinder the safe and long-term operation of nuclear power plants reliant on safety-critical concrete infrastructure [25–29]. A proper understanding of deterioration and ageing mechanisms requires a research strategy based on combined experimental and theoretical studies, following a multidisciplinary approach, and utilizing state-of-the-art experimental and modelling techniques. Material characterization at different length scales is necessary, focusing on the physical understanding of the degradation processes (e.g. neutron and gamma radiation, internal swelling reactions, liner corrosion, etc.) as well as physical phenomena (drying, creep, shrinkage, etc.), and their influence on macroscopic mechanical properties and structural/functional integrity of the components.

Although the ACES project states that the focus is on Gen-II and III reactors, in reality since the R&D development is on the ageing of reinforced concrete it can be applicable to most reactor designs that contain reinforced concrete structures in normal loading conditions (normal ageing and effect of radiation), so also applicable to GIF. Therefore, the ACES project aims at having a significant impact on the safety of operational Gen-II and III NPPs and impacting the design of next-generation



**Fig. 5.** A sketch to illustrate some basic design principles of reactor core of SCW-SMR: (a) top view of the reactor core; top cover removed; (b) side view onto the head end of the reactor core; front plate and outer cylinder of the reflector removed [22].

plants. ACES will improve the understanding of the ageing/deterioration of concrete and will demonstrate and quantify inherent safety margins introduced by the conservative approaches used during design and defined by codes and standards employed throughout the life of the plant. The outcomes from ACES will therefore support the long-term operation of NPPs. This will be achieved by using more advanced and realistic scientific methods to assess the integrity of NPP concrete infrastructure. ACES provides evidence to support the methods by carrying out various tests, including large-scale tests based on replicated scenarios of NPPs. This is being achieved by the development and validation of [30]:

- critical review of ageing management practice across EU nuclear power plants focusing on deterioration and ageing mechanisms of reinforced concrete, linked to decision-based assessment criteria.
- Improved engineering methods to assess components under long-term operation taking into account specific operational demands.
- Integrated probabilistic assessment methods to account for uncertainties and improve inspection capabilities.
- Innovative quantitative methodologies to transfer laboratory material properties to assess the structural integrity of large concrete components.
- Advanced simulation tools based on nonlinear finite element method, lattice-based models, FFT-based simulation, and combined empirical and mathematical material models reflecting the latest finding related to the combined effect of various loading (e.g. environment, radiation, internal swelling reactions) on concrete performance, i.e., the creation and evolution of deterioration areas in concrete structures.
- Improved understanding of internal swelling reactions and temperature/moisture effects on the delayed strains of containment buildings; improved prediction of the evolution of moisture, strain and stress of prestressed concrete containment buildings, during operational and accidental phases.
- Improved assessment of the effects of prolonged irradiation of the concrete biological shield using a holistic approach combining operating conditions, materials degradation and structural significance.
- Improved understanding of corrosion phenomena focusing on embedded liners, predicting the occurrence of

corrosion and developing an innovative inspection tool for early detection of corrosion.

Furthermore, an important goal of ACES is the dissemination and education of the nuclear engineering and research community. This will improve and harmonise knowledge about NPPs ageing and thereby ensure a high impact on project results. The education will be carried out through workshops and training organised during the project. ACES results will be disseminated to the main European and international stakeholders.

The ACES project consists of seven work packages. Figure 6 illustrates the overall structure of the ACES work plan. The ACES work packages are WP1 – state-of-the-art of quantitative assessment of ageing of concrete SSC in NPPs; WP2 – corrosion assessment of embedded liners in concrete; WP3 – characterization, prediction and monitoring of internal swelling reactions in concrete; WP4 – delayed strains of containment buildings in operational and accidental conditions; WP5 – assessing the performance of irradiated concrete; WP6 – dissemination, communication and training; and, WP7 – project management.

WP1 provides an overview of existing approaches for quantitative assessment of concrete performance and identifies major gaps in technical competencies and scientific knowledge for new approaches. WP2–WP5 are dealing with technical aspects (experimental and modelling) and aim at answering industrial key issues as described before. WP6 addresses activities for dissemination and exploitation of results and training as well as End-User engagement. WP7 is dedicated to project strategic and operational management. WP7 provides the overall internal management of ACES, and the administrative control including liaison with the EC.

ACES engages 11 partners from five EU Member States (VTT, SCK CEN, ZAG, IRSN, EDF, CTU, CVR, ENGIE LAB and CEA) and two non-EU countries (ENERGORISK, ORNL). The core consortium was born within the ICIC (International Committee on Irradiated Concrete), but then grew with the need to address several different ageing mechanisms. Specific partners were invited to join and contribute to the project in their field of expertise to complement the competence of the consortium. An American partner was added due to the world-leading excellence brought by this partner (came with its

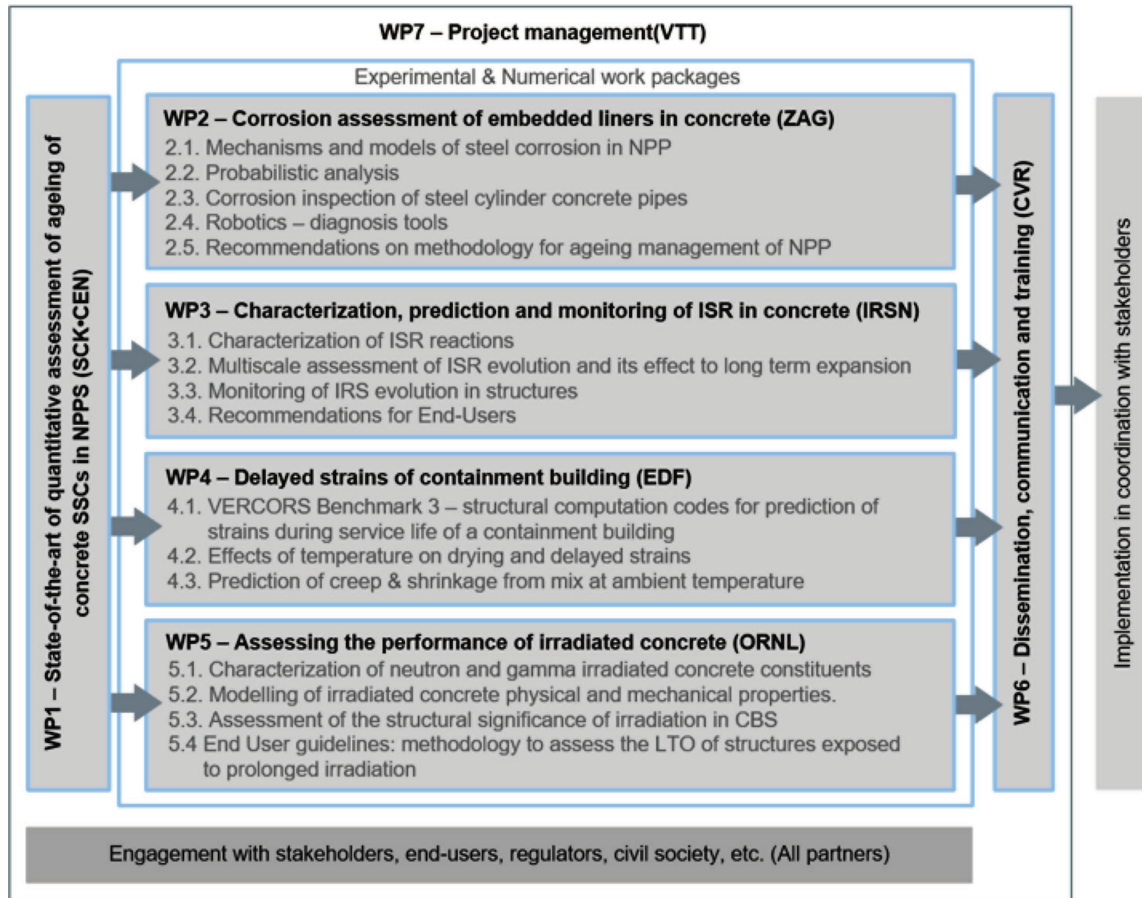


Fig. 6. Overall structure of the ACES work plan.

own funding). Many additional organizations could have joined but unfortunately, the size of the budget limited its numbers. However, most of these interested partners have joined the End Users Group to contribute through collaboration.

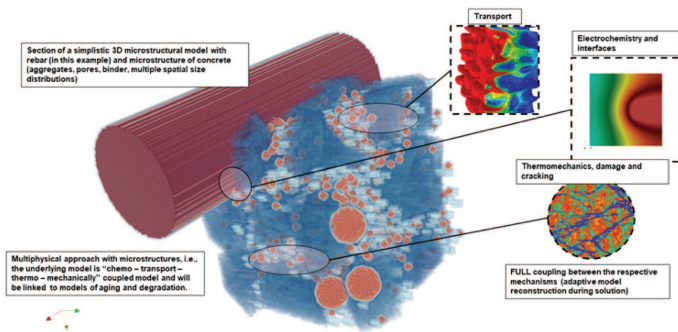
## 5.2 Key results

The key results of the ACES project include:

- a focused review of the current state-of-the-art of ageing processes of reinforced concrete SSCs in the framework of long-term operation of NPPs, including the effect of long-term climate change on environmental loading with the potential impact on concrete performance, and the synergetic effects of ageing processes under single and multiple loading conditions [31]. ACES will identify how the integration of monitoring techniques, experimental data, ageing models and modelling techniques can help to establish quantitative criteria for concrete performance in terms of the current condition, the evolution of the concrete condition, and consequences for safety.
- An improvement of phenomenological understanding and optimization of earlier detection of corrosion degradation, specifically related to the special conditions

leading to chloride-induced corrosion of steel cylinder concrete pipes and crevice corrosion of steel liner embedded in the concrete of containment building (Fig. 7). Furthermore, a robotic platform will be developed with the ability to handle and applied suitable NDT and electrochemical techniques for corrosion inspection on the internal side of steel cylinder concrete pipes will be performed (Fig. 8).

- A contribution to the assessment of concrete structures affected by internal swelling reactions by investigating the interaction mechanisms between internal swelling reactions and other ageing phenomena such as creep and shrinkage. Furthermore, ACES will develop methodologies for extrapolating the long-term concrete degradation based on common civil engineering practices and in-situ monitoring, and also by developing robust and advanced prediction tools allowing the simulation of the long-term behaviour of NPP affected by internal swelling reactions based on a multi-scale approach.
- A validation of existing constitutive laws and structural modelling approaches regarding the simulation of containment behaviour during operational phases. Furthermore, contributed to the largest database on a single concrete mix with drying, creep and shrinkage behaviour at ambient and elevated temperatures (up to 150 °C). Also, a new empirical model for concrete



**Fig. 7.** Phenomenological modelling concept of steel corrosion.

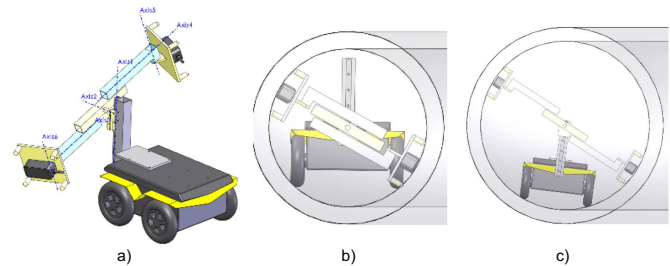
behaviour up to 150 °C is proposed as well as evolutions of the FIB Code Model to better account for multi-axial effects on creep at moderate temperatures (up to 40 °C). Finally, the validation of existing models for the prediction of concrete delayed strains based on concrete composition.

- Generating critical data currently missing from the open literature on the neutron-irradiation-induced degradation of concrete aggregates relevant for European NPPs. Collecting data from in-service and accelerated irradiated concrete is key to understanding the significance of accelerated irradiation conditions against LWRs operating conditions. Furthermore, a rigorous benchmark of a large variety of modelling approaches, the capabilities and limitations of each model will be assessed and lead to thoroughly validated irradiated concrete models that are expected to serve as references for future uses by industry, regulatory bodies and code activities. The main advance of this research will be to establish a rigorous structural assessment based on operation data, materials consideration and validated materials and structural models.

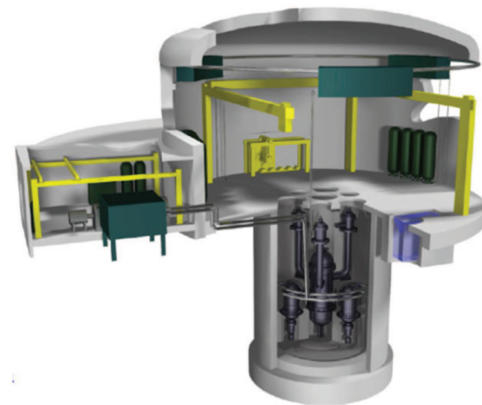
### 5.3 Recommendations for the future

ACES will have a significant impact on the safety of nuclear power plants by demonstrating and revealing inherent safety margins being introduced by the conservative approaches used during design and being dictated by codes and standards used through the lifetime of the plant. The outcome of ACES will support the long-term operation of nuclear power plants, by using more realistic approaches for the integrity assessment of reinforced concrete SCC's and provide evidence by large-scale tests. The results of ACES will be disseminated in key industry circles. The results will be presented to key industry players such as UR NRC which also chairs the ACI 349, and therefore can bridge the integration of the new knowledge into standardization committees and initiate discuss how to integrate it.

Through the innovative methods and tools developed and the integration of knowledge carried out in ACES, the project results will allow companies to increase the reliability of safety measurements in nuclear power plants. ACES will contribute to the safe operation of nuclear



**Fig. 8.** (a) Concept of robotic solution design, (b) and (c) its deployment in a pipe with regard to the diameter.



**Fig. 9.** ALLEGRO reactor system.

power plants and to maintaining high safety standards that can be realised because of more accurate assessment methods resulting in the demonstration of an increased margin of safety. Through proactive management of NPPs ageing and long-term operation, the NPP operators will decrease costs related to better planning of the suspension of NPP activity for maintenance and to the reduction of the inspection time. This will in turn strengthen their competitiveness in the global energy market by replying in a cost-effective way to the increasing demand for nuclear power, foreseen by the International Energy Agency for 2050.

## 6 SafeG: Safety of GFR Through Innovative Materials, Technologies and Processes

### 6.1 Key objectives

The safety of the GFR demonstrator ALLEGRO (see Fig. 9) is to be enhanced through the use of innovative technologies, and materials and using unique know-how that has been built both inside and outside Europe over the last 20 years. The most important areas of ALLEGRO safety improvements tackled by the SafeG project are:

To solve remaining open questions in residual heat removal in accident conditions, leading to practical elimination of severe accidents, through the innovative design of the reactor core, diversified ways of passive reac-

tor shutdown, passive decay heat removal systems, and instrumentation.

To strengthen the inherent safety of the key reactor components by review of obsolete material and technologies reference options, selection of innovative options, and designs based on these innovative options.

Review the GFR reference options [32] in materials and technologies, using experience gained in national research programs, the know-how of the consortium and stakeholders, and experience from the operation of various research facilities and high-temperature nuclear reactors. The aim is to increase the inherent safety of GFRs.

Adapting GFR safety to changing needs in electricity consumption worldwide with an increased and decentralized portion of nuclear electricity by the study of various fuel cycles and their suitability from the safety and proliferation resistance points of view.

Boosting interest in GFR research by wide involvement of universities, promotion of GFR-oriented topics of master theses and dissertations, organizing topical workshops including hands-on training and on-job training connected with staff exchange.

Deepen collaboration with international non-EU research teams (Kyoto University, University of Cambridge, University of Sheffield), relevant European and international bodies (GIF, standardization bodies) and partners with experience/interest in GFR.

## 6.2 Key results

The project is currently in its initial phase. The global objective of the SafeG project is to develop the gas-cooled fast reactor technology and strengthen its safety by solving open issues concerning the GFR technology and its demonstrator ALLEGRO. The project is divided into 7 Work Packages, four of them dealing with open research and development problems of GFRs, namely the core safety and proliferation resistance (WP1), advanced materials and technologies (WP2), decay heat removal (WP3), standardization and codes (WP4). Much of the effort is dedicated to education and training activities sheltered by WP5. Dissemination and outreach activities are included in WP6 while WP7 ensures smooth management and execution of the project.

The main goal of WP1 is to determine the driver and refractory cores satisfying both the performance and safety requirements. In order to achieve this double goal in parallel, the necessary multi-discipline approach will be applied in an iterative way. Namely, neutronics, thermal-hydraulics, and thermal mechanics codes are to be used in the analyses. The normal operation power limitations – both global and local, the maximum burnup, and the core size are the key parameters of the above multi-discipline core design. The main goal of these activities is to optimize the existing designs of the driver and refractory cores from the point of view of safety, related mostly to core designs with more favourable behaviour in unprotected transient scenarios.

One of the main goals of WP2 is to implement materials with better performance for the primary circuit. The process of selection and evaluation of materials for internals will begin with the identification of potential candidate materials based on desired material properties. In general, materials that have been used in High-Temperature Reactors (HTR), and new materials such as SiC-based composites and High Entropy Alloys (HEA) will be researched. This activity will generate unique experimental data of material properties of these advanced materials in various GFR-specific conditions like exposing them to He-N<sub>2</sub> mixture at very high temperatures, which will be beneficial not only for the project and GFR research, but will be included in the existing codes for reactor development.

Concerning WP3 the various phenomena of decay heat removal in GFRs will be simulated both numerically and experimentally. For numerical simulations, existing, but correspondingly upgraded models of DHR systems for the CATHARE, RELAP5 and MELCOR codes will be used, with support of CFD simulation for selected separated effects study. Multiple codes are used for detailed analysis of decay heat removal phenomena in the proposed concept. The ambition is to finish the fully passive decay heat removal strategy for ALLEGRO. To achieve this goal, detailed studies of thermal-hydraulics phenomena affecting the decay heat removal will be performed, complemented by designing the fully passive decay heat removal systems, and experiments studying decay heat removal performed at state-of-the-art GFR-dedicated facility S-ALLEGRO.

The results and experience from WP1 to WP3 will be collected in WP4 and evaluated, with emphasis on areas where an insufficient amount of solid data will be identified. The qualification options, and options for qualification of new fuels for GFRs will be explored, finishing with a proposal of a fuel qualification plan.

The reference GFR core should employ the high-temperature resistant cladding materials (SiC) as well as the UPuC (or UC) fuel pellet matrix with the possibility to close the fuel cycle. It was envisaged that such fuel assemblies will be qualified also in the ALLEGRO cores specified in the present project (UO<sub>2</sub> and MOX, refractory cores). All the mentioned cores will be important for the fuel qualification process. However, recent advances introduced new variables, which must be considered. The rapid development of the Accident Tolerant Claddings for LWRs increased the knowledge base significantly, mainly with respect to the manufacturing a joining of the SiC–SiC tubes. The advanced multiscale modelling and PIE techniques may allow, to some extent relevant to GFR technology, to leverage the experimental results which will be produced in the next few years in the thermal neutron spectrum.

The fuel qualification plan will be done by taking into account all the above-mentioned variables and using the broad experience of the consortium members in fuel development, qualification and safety assessment.

The main driving force for research and development within this project will be the students and early career researchers employed directly through the project funds or

contributing indirectly through complementary research activities in participating institutions. It is envisaged that a PhD. students involved in the project will spend a significant proportion of their research time in other than their home institutions, learning state-of-the-art methods and acquiring skills from other institutions participating in the project.

Participating universities will share their experience in teaching advanced nuclear technology curricula in their respective institutions with subsequent implementation of best practices. It is also envisaged that senior academics and post-doctoral researchers engaged in the project will offer and supervise a number of master-level theses on topics relevant to GFR technology development, therefore vastly amplifying the amount and value of research conducted within the project which is directly sponsored by the EC.

Dissemination of knowledge amongst the project participants will also be accomplished through regular face-to-face project meetings, dedicated thematic workshops and a summer school – all aiming at creating a self-sustainable and robust international network of experts and researchers at different stages of their careers engaged in the development of GFR technology, exchange of ideas to promote technological innovation, and information exchange to accelerate the development and eventual deployment of GFRs.

### 6.3 Recommendations for the future

The SafeG project focuses on the introduction of the GFR technology as a commercial, industrial source of both electricity and high-temperature heat for industrial applications. Nuclear power technology is a sustainable, low-carbon technology. It is an efficient alternative to fossil energy sources with a significant reduction of CO<sub>2</sub> emissions. Moreover, GFR technology can become a source of high-potential heat for the industry. One of its promising applications is to produce hydrogen, being considered the fuel of the future, reducing, even more, the carbon footprint of human activities.

ALLEGRO, as a demonstrator of Generation-IV GFR technology, will use a fast neutron spectrum, which enables re-using of already available fissile and fertile materials (including depleted uranium from enrichment plants). Through the combination of a fast-neutron spectrum and full recycling of actinides, the GFR reactors will minimize the production of long-lived radioactive waste isotopes, with a significant positive environmental impact.

## 7 Conclusion

The paper briefly presents five Euratom projects started since late 2017 in support of the R&D of four Generation-IV reactor systems: Sodium Fast Reactor, Molten Salt Fast Reactor, Supercritical Water Cooled Reactor and Gas Cooled Fast Reactor as well as cross-cutting activities on safety-critical concrete infrastructure.

On one hand, the **ESFR-SMART** project continues the development of the European Sodium Fast Reactor concept following up the EFR and CP ESFR projects, especially in terms of safety enhancement and design simplification. On the other hand, R&D activities in support of the Sodium Fast Reactors, in general, are performed in terms of codes validation and calibration, new experiments and new instrumentation, support of sodium facilities and measurements of MOX fuel properties. The project is ongoing and scheduled to finish in August 2022.

The **SAMOSAFER** project aims at the modelling of MSR including as many of the MSR-specific physics and chemistry as possible and at the experimental validation of these models. We build upon the expertise and experience generated in the SAMOFAR project and extend these to ensure the MSR will comply with all future regulations. The SAMOSAFER project is well on track and is expected to finish in 2023.

The **ECC-SMART** project benefiting from the European, Chinese, Canadian and Ukraine synergies helps to move forward the development of the nuclear supercritical water-cooled technology by preparing the design requirements and recommendations for the licensing of the future small modular reactor cooled supercritical water (SCW-SMR). The project started in September 2020 and is close to half with a scheduled end in August 2024. Most of the planned experiments have been started and the first results are evaluated and continuously published. Based on the preliminary results from neutronic physics and core layout, some modifications have been proposed and they should be included in the design concept. Despite the pandemic restrictions and inability to hold face-to-face meetings, the project is progressing well and the work activities in each work package are properly performed.

The research approach undertaken in the **ACES** project supports the development of a physical understanding of concrete degradation processes and their influence on macroscopic mechanical properties and structural/functional integrity of concrete structures. The research addresses important knowledge gaps such as (i) the corrosion of liners embedded in concrete; (ii) internal swelling reactions of concrete; (iii) the drying, creep and shrinkage of the containment structure; and (iv) the effects of prolonged irradiation of CBS. When completed in 2024, the ACES project will have developed industry guidelines enabling End Users to assess the susceptibility and structural assessment of their structures subject to the different deterioration and ageing mechanisms under consideration.

The **SafeG** project is a part of initiatives leading to the construction of the Gas-cooled Fast Reactor (GFR) experimental reactor ALLEGRO. The project's approach is based on the long-term and extensive experience of the project consortium partners in nuclear R&D, development and application of various simulation tools, licensing of nuclear installations, material science and reactor safety in general. The global objective of the SafeG project is to further develop the gas-cooled fast reactor technology and strengthen its safety by solving open issues concerning the GFR technology and its demonstrator ALLEGRO.

## Conflict of interests

The authors declare that they have no competing interests to report.

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## Data availability statement

Data associated with this article are disclosed according to the Consortium Agreements of every project.

## Author contribution statement

Sections 1 and 2 was written by K. Mikityuk, and Section 3 by J.-L. Kloosterman, Section 4 by M. Šířpová, Section 5 by M. Ferreira, Section 6 by B. Hatala, and Section 7 was written by all co-authors.

## References

1. M. Schyns, R. Adinolfi, N. Camarcat, J. Duspiva, M. Frignani, J.-C. Garnier, J.-M. Hamy, ESNII Vision Paper, 2021 Ed., ISBN 978-2-919313-27-3.
2. Special Issue: EU ESRF-SMART Project, ASME J. of Nuclear Rad. Sci. **8**, 1 (January 2022). <https://asmedigitalcollection.asme.org/nuclearengineering/issue/8/1>.
3. S. Mambelli, Analytical and experimental study of chugging boiling instability: The CHUG project, MS thesis, ETH Zurich. July 2018. <https://doi.org/10.5281/zenodo.1311464>.
4. A. Lecoanet, M. Gradeck, X. Gaus-Liu, T. Cron, B. Fluhrer, F. Payot, C. Journeau, N. Rimbart, Ablation of a solid material by high-temperature liquid jet impingement: An application to corium jet impingement on a sodium fast reactor core-catcher, ASME J. of Nuclear Rad. Sci. **8**, 011308 (2022).
5. N. Krauter, V. Galindo, T. Wondrak, S. Eckert, G. Gerbeth, Eddy current flow meter performance in liquid metal flows inclined to the sensor axis, ASME J. of Nuclear Rad. Sci. **8**, 011303 (2022).
6. A. Ponomarev, K. Mikityuk, L. Zhang, E. Nikitin, E. Fridman, F. Alvarez-Velarde, P. Romojaro Otero, A. Jiménez-Carrascosa, N. García-Herranz, B. Lindley, U. Baker, A. Seubert, R. Henry, Superphénix benchmark Part I: Results of static neutronics, ASME J. of Nuclear Rad. Sci. **8**, 011320 (2022).
7. A. Ponomarev, K. Mikityuk, E. Fridman, V. A. Di Nora, E. Bubelis, M. Schikorr, “Superphénix Benchmark Part II: Transient Results”, ASME J. of Nuclear Rad. Sci. **8**, 011321 (2022).
8. S. Perez-Martin, M. Anderhuber, L. Laborde, N. Girault, C. Lombardo, L. Ammirabile, K. Mikityuk, S. Mimouni, C. Péniguel, W. Pfrang, Evaluation of sodium boiling models using KNS-37 loss of flow experiments, ASME J. of Nuclear Rad. Sci. **8**, 011310 (2022).
9. A. Rineiski, C. Mériot, M. Marchetti, J. Krepel, C. Coquelet-Pascal, H. Tsige-Tamirat, F. Álvarez-Velarde, E. Girardi, K. Mikityuk, ESRF-SMART core safety measures and their preliminary assessment, ASME J. of Nuclear Rad. Sci. **8**, 011322 (2022).
10. J. Guidez, J. Bodi, K. Mikityuk, E. Girardi, B. Carlucci, New reactor safety measures for the European sodium fast reactor – Part I: Conceptual design, ASME J. of Nuclear Rad. Sci. **8**, 011311 (2022).
11. E. Fridman, F. Álvarez Velarde, P. Romojaro Otero, H. Tsige-Tamirat, A. Jiménez Carrascosa, N. García Herranz, F. Bernard, R. Gregg, U. Davies, J. Krepel, B. Lindley, S. Massara, S. Pomerouly, E. Girardi, K. Mikityuk, Neutronic analysis of the European sodium fast reactor: Part II – Burnup results, ASME J. of Nuclear Rad. Sci. **8**, 011301 (2022).
12. J.S. Narváez Arrúa, A. Cammi, S. Lorenzi, Numerical methodology for design and evaluation of natural circulation systems for MSR applications, in submitted to *ANS Annual Meeting* (Anaheim, US, 12–16 June 2022).
13. A. Di Ronco, S. Lorenzi, F. Giacobbo, A. Cammi, Multiphysics analysis of RANS-based turbulent transport of solid fission products in the Molten Salt Fast Reactor. submitted to *Nucl. Eng. Design* **391**, 111739 (2022).
14. J. Diet, J. Krepel, S. Nichenko, MSR fuel cycle and thermo-dynamic simulations, in submitted to *FR22* (Austria, Vienna, 19–22 April 2022).
15. F. Caruggi, A. Cammi, E. Cervi, A. Di Ronco, S. Lorenzi, Multiphysics modelling of gaseous fission products removal in the molten salt fast reactor, submitted to *Ann. Nucl. Energy* (2022).
16. B.J. Kaaks, J.W.A. Reus, M. Rohde, J.L. Kloosterman, D. Lathouwers, Numerical study of phase-change phenomena: A conservative linearized enthalpy approach, in *19th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-19)* (Brussels, Belgium, 6–11 March 2022).
17. F. Caruggi, A. Cammi, E. Cervi, A. Di Ronco, S. Lorenzi, Modelling and simulation of the gaseous fission product removal in the molten salt fast reactor, in *19th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH 19)* (Brussels, Belgium, 6–11 March 2022).
18. S. Nichenko, L. Terttaliisa, J. Kalilainen, MSR simulation with cGEMS: Fission product release and aerosol formation, in *Cooperative Severe Accident Research Program (CSARP)* (7–11 June, 2021).
19. A. Di Ronco, S. Lorenzi, F. Giacobbo, A. Cammi, An Eulerian single-phase transport model for solid fission products in the molten salt fast reactor: development of an analytical solution for verification purposes, *Front. Energy Res.*, **9**, (29 June 2021). *Sec. Nuclear Energy*, <https://doi.org/10.3389/fenrg.2021.692627>.
20. G. Merla, A. Cammi, S. Lorenzi, A new reactivity control approach for circulating fuel reactors, in *NENE 2021* (Slovenia, Bled, 6–9 September 2021).
21. M. Santanoceto, M. Tibergera, Z. Perkó, S. Dulla, D. Lathouwers, Preliminary uncertainty and sensitivity analysis of the Molten Salt Fast Reactor steady-state using a Polynomial Chaos Expansion method, *Ann. Nucl. Energy* **159**, 108311 (2021).
22. T. Schulenberg, I. Otic, Suggestion for design of a Small Modular SCWR, in *Proceedings of the 10th International*

- Symposium on SCWRs (ISSCWR-10)* (Prague, the Czech Republic, March 15–19, 2021) p. 17.
23. Y. Dubyk, V. Filonov, O. Kovalenko, Y. Filonova, Deteriorated heat transfer influence on the stress-strain state of SMR SCWR fuel bundles, *ASME J of Nuclear Rad. Sci.* **8**, 031105 (2022).
  24. A. Pucciarelli, S. Kassem, W. Ambrosini, Overview of a theory for planning similar experiments with different fluids at supercritical pressure, *Energies*, MDPI **14**, 1–22 (June 2021). <https://doi.org/10.3390/en14123695>.
  25. IAEA, Ageing management of concrete structures in NPP (IAEA, Nuclear Energy Series, NP-T-3.5, 2016) p. 372.
  26. IAEA, Ageing Management for NPP (IAEA, Safety Guide, NS-G-2.12, 2009) p. 65.
  27. U.S.NRC, Expanded materials degradation assessment (EMDA) – Volume 4: Ageing of concrete and civil structures. USNRC, NUREG-CR-7153, Vol. 4, ORNL/TM-2013/532, 2014, p. 135.
  28. IAEA, Ageing Management and Development of a Programme for LTO of NPP (IAEA, Safety Guide, SSG-48, 2018) p. 64.
  29. IAEA, Periodic Safety Review for Nuclear Power Plants (IAEA, Safety Standards Series, SSG-25) 2013.
  30. ACES H2020 Project, Towards improved assessment of safety performance for LTO of nuclear civil engineering structures, Description of Action (ACES Project, 2020), p. 70.
  31. D. Jacques, L. Yu, M. Ferreira, T. Oey, Overview of state-of-the-art knowledge for the quantitative assessment of the ageing/deterioration of concrete in nuclear power plant systems, structures, and components, ACES Project Deliverable D1.1, 2021.09.27, p. 217.
  32. C. Poette, ALLEGRO preliminary viability Report, CEA/DEN/CAD/DER/SESI/LCSI NT-DO12, December 2009.

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