Symposium on ENEA ReseArch For Innovative Nuclear application

SERAFIN WORKSHOP

WORKSHOP PROCEEDINGS

17th – 18th June, 2025,
Faculty of Engineering,
University of Bologna
Viale del Risorgimento 2,
40136, Bologna, Italy

Symposium on ENEA ReseArch
For Innovative Nuclear application

SERAFIN WORKSHOP

This document collects the scientific contributions presented at the SERAFIN 2025 workshop and represents a joint effort of researchers from research institutions, universities, and industry. The workshop was organized and coordinated by the **ENEA Nuclear Energy Systems Division**, with contributions from both academic and industrial experts.

The introduction of the volume is written by **Mariano Tarantino**,
Head of the Nuclear Energy Systems Division (ENEA).

The full list of authors is provided within the individual contributions.

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2026 ENEA
Italian National Agency for New Technologies, Energy and Sustainable Economic Development

The workshop was organized and coordinated by the ENEA Nuclear Energy Systems Division, with contributions from both academic and industrial expert.

The introduction of the volume is written by Mariano Tarantino, Head of the Nuclear Energy Systems Division (ENEA)

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FOREWORD

In the Italian context, the debate on nuclear technology as a potential contributor to energy sustainability and to the achievement of net-zero emissions by 2050 has recently regained momentum. Within this framework, ENEA, and in particular the Nuclear Energy Systems Division of the Nuclear Department (NUC-ENER), reaffirms its role as a national reference point in nuclear research and innovation.

The NUC-ENER Division develops advanced expertise in the design, analysis, and technological development of nuclear systems for energy applications, addressing both fission and fusion technologies. By supporting institutions, industry, and society, the Division contributes to the safety, sustainability, and design of next-generation nuclear reactors, including Light Water Small Modular Reactors (LW-SMRs) and Generation IV systems, while promoting technology transfer and active participation in national and international research networks.

The first workshop edition ("Critical Mass for Synergy") was held from 12 to 14 July 2023 at the ENEA Bologna Research Centre and was jointly organized by the FSN-SICNUC Division and FSN-PROIN Section, subsequently restructured into the NUC-ENER Division. The second workshop edition, entitled *Symposium on ENEA ReseArch For Innovative Nuclear application (SERAFIN)*, took place from 17 to 18 June 2025 and was organized by the NUC-ENER Division with the support of the University of Bologna to strengthen internal collaboration and showcase ENEA's research excellence in the nuclear field. The workshop was structured into four thematic sessions:

- *Safety and Risk Assessment in Nuclear Installations*
- *Innovative Design and Modelling of Nuclear Systems*
- *Advanced Technologies for Future Nuclear Applications*
- *Experimental Campaigns and Facility Updates*

In addition, two roundtables addressed key cross-cutting issues:

- *Severe Accident Analysis for Advanced Reactors: Safety-by-Design, Mitigation Strategies, and Regulatory Challenges*
- *Experimental Data for Innovative Nuclear Reactors*

The SERAFIN Workshop provided a valuable opportunity for researchers, academics, PhD candidates, and students to exchange knowledge, discuss recent advances, and contribute to the development of innovative nuclear technologies in support of Italy's energy transition.

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GROUP PHOTOS

CONFERENCE PROGRAM

17th – 18th June, 2025, Faculty of Engineering, University of Bologna, Italy

Tuesday, June 17th, 2025

08:30 Registration

09:00 Opening remarks

Alessandro Dodaro – Head of Nuclear Department

Mariano Tarantino – Head of ENEA NUC-ENER Division

Matteo Gherardi – Associate Professor, University of Bologna

09:15 **Session 1: Safety and Risk Assessment in Nuclear Installations**
Chair: Simone Gianfelici

Deterministic Analyses to Support Small Modular Reactor Safety Demonstration
Fulvio Mascari, ENEA

Integral effect preliminary tests in CIRCE-THETIS facility: heat losses characterization and transition from forced to natural circulation experiments
Alessandro Bellomo, University of Pisa

Application of the REPAS methodology to assess the reliability of the EHRS in a generic iPWR SMR
Erik Cilia, University of Bologna

Supporting the ESFR-SMR Design: Safety Analyses and Fuel Optimization Activities at ENEA
Massimiliano Polidori, ENEA

Q&A

10:30 Poster Session - Coffee Break

11:00 Italy and Advanced Nuclear Technologies: Insights and Recommendations from the PNNS Working Group on Nuclear Fission
Gianluca Benamati, ENEA

11:30 Roundtable: Severe Accident Analysis for Advanced Reactors
Chair: Fulvio Mascari

12:30 Group photo and Lunch

14:00 **Session 2: Innovative Design and Modelling of Nuclear Systems**
Chair: Patrizio Console Camprini

The IAEA Coordinated Research Project on NACIE-UP facility: outcome of ENEA activities
Roberto Da Vià, ENEA

From Needs to Development, Validation and Application (Needs): The Virtuous Cycle for LFR Core Design
Giacomo Grasso, ENEA

Enabling advanced thermal-hydraulics knowledge: the NETT1 Laboratory
Francesco Lodi, ENEA

Enhancing LFR Analysis Accuracy: Refining Nuclear Data by Differential and Integral Measurements
Donato Maurizio Castelluccio, ENEA

Q&A

15:30 Poster Session - Coffee Break

16:00 **Session 3: Advanced Technologies for Future Nuclear Applications**
Chair: Barbara Ferrucci

MCNP simulation of the Intra-Operative Radiation Therapy (IORT) treatment of solid cancers with fast neutrons and comparison with the dosimetry parameters of standard IORT techniques adopting X-rays and electrons
Massimo Sarotto, ENEA

The SELENE project: the Italian contribution to a lunar base
Roberto Pergreffi, ENEA

Design methods for Beam Intercepting Devices for Particle Accelerators
Luca Tricarico, University of Bologna

Improving Initialization of ASTEC's Thermal-Hydraulic Solver with Machine Learning-Based Methods for Enhanced Convergence in Severe Accident Simulations
Marcello Savini, University of Bologna

Q&A

Wednesday, June 18th, 2025

08:30 Registration

09:00 Data management, Nuclear Anthologies and CONNECT-NM project

Emanuele Ghedini, University of Bologna

09:30 **Session 4: Experimental Campaigns and Facility Updates**
Chair: Carlo Carelli

Flow Induced Vibration (FIV) experiment in the HELENA facility
Ivan Di Piazza, ENEA

Lead corrosion performance of Al and Cr based pack cementation coatings on 316Ti without pre-oxidation treatment
Chantal Vannini, *new*cleo

NACIE-LHT revamping and experimental test
Pietro Cioli Puviani, *new*cleo

Measurement of ^{63,65}Cu neutron capture cross sections at the n_TOF facility
Nicholas Pieretti, University of Bologna

Q&A

10:45 Poster Session - Coffee Break

11:15 Roundtable: Experimental data for innovative nuclear reactors
Chair: Marco Utili

12:15 Conclusion and Final Remarks

Mariano Tarantino – Head of ENEA NUC-ENER Division

Matteo Gherardi – Associate Professor, University of Bologna

12:30 End of Workshop

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INTRODUCTION

Following the experience of the first edition, the second NUC-ENER Workshop was conceived with a renewed focus on external outreach. The main objective of this edition was to present, in a structured and comprehensive manner, the full spectrum of research activities carried out within the ENEA Nuclear Energy Systems Division of the Nuclear Department.

The workshop aimed to highlight the Division's multidisciplinary expertise, ongoing research lines, experimental capabilities, and technological developments, fostering dialogue with universities, research institutions, and industry stakeholders. Through thematic sessions and dedicated discussions, the event provided a clear overview of ENEA's role and contributions in the field of innovative nuclear technologies.

This document collects the extended abstracts presented during the workshop and reflects the scientific breadth and relevance of the activities discussed.

Mariano Tarantino

Head of the ENEA NUC-ENER Division

SESSION 1

SAFETY AND RISK ASSESSMENT IN NUCLEAR INSTALLATIONS

Chairperson: *S. Gianfelici*

Session 1 was aimed at providing an overview of a set of activities addressing one of the core and longstanding lines of action characterizing the ENEA laboratories involved in nuclear safety, namely the assessment, demonstration, and enhancement of the safety of nuclear installations through advanced analytical and experimental approaches.

Most of the activities presented during the session are framed within institutional research programmes, national and international projects, and collaborations with academia, and are primarily focused on supporting the development, design, and assessment of innovative nuclear reactor concepts, with particular attention to Small Modular Reactors and Generation IV systems. The contributions provided an overview of methodologies and tools spanning from deterministic safety analyses to risk-informed approaches, highlighting the integration of system thermal-hydraulics, severe accident modelling, and reliability assessment of safety systems.

Several presentations emphasized the role of advanced deterministic analyses in supporting safety demonstrations, including the modelling of design-basis and beyond-design-basis accident scenarios and the assessment of decay heat removal capabilities under natural circulation conditions. In this context, experimental activities were also illustrated, aimed at reproducing integral thermal-hydraulic phenomena and providing experimental evidence for the validation of analytical models and the characterization of safety-related components and systems.

Other contributions highlighted the importance of probabilistic and hybrid methodologies for safety and risk assessment. The application of structured approaches combining deterministic simulations with probabilistic techniques, such as Probabilistic Safety Assessment and reliability-oriented methodologies, was presented as an essential tool to quantify uncertainties, identify potential functional failures of passive systems, and support risk-informed decision-making in reactor design and safety evaluation.

Overall, the session underlined the relevance of combining continuous methodological development, advanced numerical tools, and experimental validation to maintain a high level of competence in the field of nuclear safety. The presented activities clearly demonstrated the role of ENEA and its partners in delivering qualified technical support to research, industry, and regulatory-oriented activities, and in contributing to the advancement of safety assessment methodologies for present and future nuclear installations.

DETERMINISTIC ANALYSES TO SUPPORT SMALL MODULAR REACTOR SAFETY DEMONSTRATION

F. Mascari[1*], C. Lombardo[1], M. Polidori[1], G. Grippo[2], E. Cilia[2], M. Savini[2], G. Agnello[3], F. Rocchi[1]

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Abstract

Safety is the key pillar for the deployment of nuclear technology. Considering the performance of the different Nuclear Power Plants (NPP) in operation around the world, the current generation of large water reactors has proven to be a reliable and safe technology. Still, people and policymakers are requesting nuclear technologies with increased inherent safety to strengthen the contribution of nuclear energy to the overall energy mix, in line with the 2050 net-zero CO_2 objective. In this framework, reactors using passive mitigation strategies have become a flagship of the nuclear industry, which is striving to incorporate these features into advanced designs. Deterministic Safety Analyses (DSAs) are part of the Defence-in-Depth (DiD) approach to demonstrate the safety of nuclear technologies and designs and play a crucial role in characterizing a plant under accident conditions. They describe its time-dependent behavior and identify key transient phenomena and processes, as well as the methodologies used to demonstrate that safety limits are respected. Several activities have been developed in the NUC-ENER-SIC Laboratory over the last decade to support the safety analysis of current and advanced technologies, including SMRs. These include the use and assessment of computational tools (e.g., thermal-hydraulics and severe accident codes), the development and application of methodologies for reliability assessment (e.g., REPAS) and uncertainty estimates (e.g., BEPU methodologies), the analysis of scaling issues for experimental design and code validation, and the evaluation of mitigation strategies such as In-Vessel Melt Retention (IVMR). In view of these activities, the main outcomes of ongoing efforts within various international frameworks (OECD/NEA, IAEA, EURATOM, etc.) will be presented, highlighting their contribution to the safety demonstration of both current and advanced nuclear designs.

Keywords: *DBA; Passive Systems; Safety; Severe accident; SMR*

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Introduction

Safety is the key pillar for the deployment of nuclear technology. Considering the performance of the different Nuclear Power Plants (NPP) in operation around the world, the current generation of large water reactors has proven to be a reliable and safe technology. Still, people and policymakers are requesting nuclear technologies with increased inherent safety to strengthen the contribution of nuclear energy to the overall energy mix, in line with the 2050 net-zero CO_2 objective. In this framework, reactors using passive mitigation strategies have become a flagship of the nuclear industry, which is striving to incorporate these features into advanced designs.

Deterministic Safety Analyses (DSA) is a fundamental component of the Defence-in-Depth (DiD) strategy. They play a central role in demonstrating the safety of nuclear technologies by characterizing the plant response under accident conditions, describing the time-dependent behavior of the system, and identifying the dominant physical phenomena and processes involved. These analyses are instrumental in verifying that safety limits are met, both in steady-state and transient operational conditions, and thus form the technical basis for reactor design and safety assessment. To ensure a robust safety demonstration, deterministic analyses rely on the integration of multiple elements: representative experimental data with a rigorous treatment of scaling effects, state-of-the-art computational tools, and the application of consolidated methodologies, such as the Best Estimate Plus Uncertainty (BEPU), and structured techniques to assess the reliability of passive safety systems as REliability evaluation of Passive safety Systems (REPAS).

Historically, experiments were utilized, even before the advent of computers, to estimate, understand, and prepare models phenomena that may appear in the prototype Light Water Reactors (LWRs) of various sizes and the initial and boundary conditions. Later, with the advances in NPP technology, the plant's design and operation rely more on the computer code safety analyses. Considering this the strategy to do a safety determination of reactor design and operation is to evaluate the prototype response through data from experiments, and/or computer code calculations. Due to inherent difficulties in conducting full-scale tests, in general experiments have been performed in scaled-down test facilities (Separated Effect Test Facilities – SETF-, Integral Effect Test Facilities – IETF) with the assumption that the experimental results obtained are applicable and relevant to the full-scale reference reactors. The test facilities, as well as the operating conditions, should be then properly scaled. In this context, scaling strategies must be carefully applied to ensure that the dominant physical phenomena/processes are preserved and that the results remain applicable to the reference reactor. While some scaling distortions are inevitable, they should be minimized to avoid compromising the validity of key system behaviours. Once validated against experimental data, computer codes can be used to extrapolate results to full-scale conditions. However, such extrapolations are always affected by a degree of uncertainty, stemming from both the limitations of the experimental setup and the approximations inherent in the modeling process. This underscores the essential role of experiments, not only in understanding dominant phenomena, but also in supporting the validation and uncertainty quantification of simulation tools used in safety assessments [1]. Several activities have been developed in the NUC-ENER-SIC Laboratory over the last decade to support the safety analysis of current and advanced technologies, including SMRs. These include the use and assessment of computational tools, the development and application of methodologies for reliability assessment (e.g., REPAS) and uncertainty estimates (e.g., BEPU methodologies), the analysis of scaling issues for experimental design and code validation, and the evaluation of mitigation strategies such as In-Vessel Melt Retention (IVMR). In view of these activities, the main outcomes of ongoing efforts within various international frameworks (OECD/NEA, IAEA, EURATOM, etc.) will be presented, highlighting their contribution to the safety demonstration of both current and advanced nuclear designs.

Contribution to experimental activity to develop assessment database

Among the activities carried out within the NUC-ENER-SIC Laboratory, particular emphasis should be given to the experimental work conducted at the PERSEO, HERO, and ELSMOR facilities. These experimental infrastructures have played a key role in supporting both national research initiatives and European projects focused on the development and assessment of advanced safety systems. PERSEO [2] is a full-scale SETF aimed at studying a new Passive Decay Heat Removal (DHR) system operating in natural circulation. It was built at SIET (Italy) modifying the existing PANTHERS IC-PCC facility. PERSEO was not designed to simulate a specific passive system of a particular reactor design currently in operation or under design, but its main purpose is the assessment of the performance and the efficiency of a new in-pool heat exchanger for DHR, implementing natural circulation.

In relation to HERO facility [3], ENEA contributed to its design, experimental definition, and qualification activities, supporting the development of the experimental program and the interpretation of test results. The HERO-2 facility was conceived to investigate the thermal-hydraulic behaviour of bayonet tube steam generators under natural circulation conditions, providing experimental data for the validation of system codes and for the assessment of passive decay heat removal strategies in advanced reactor concepts, including SMRs.

In relation to the ELSMOR facility [4], ENEA contributed to the development of a Phenomena Identification and Ranking Table (PIRT) to identify and characterize the most promising passive systems capable of fulfilling key safety functions in SMR designs, with particular emphasis on phenomena relevant to core heat removal by natural circulation. In addition, ENEA performed pre-test calculations using the RELAP5/Mod3.3 code to support both the experimental design and the construction of the facility. The primary objective of ELSMOR was the experimental validation of a DHR system for the European SMR (E-SMR), based on the most credible passive mechanisms for rejecting decay heat via natural circulation in a prototypical heat exchanger.

Contribution to computational tool validation

In recent years, several validation activities have been carried out within the NUC-ENER-SIC laboratory to support the qualification of computational tools for the safety analysis of current and advanced nuclear reactors. A particular emphasis has been placed on assessing the capability of system codes to simulate key phenomena relevant to SMRs, such as natural circulation, heat transfer in helical coil steam generators, the interaction between the primary system and containment, and passive system behaviour. The TRACE code has been analysed within the USNRC CAMP framework using data from the OSU-MASLWR facility. These studies demonstrated its effectiveness in modelling natural circulation under integral configurations and in predicting primary-to-secondary heat transfer and containment coupling. The associated scaling issues are also being investigated [5]. In parallel, within the PASTELS (Passive Systems: Simulating the Thermal hydraulics with Experimental Studies) project, funded by Euratom H2020, RELAP5 and CATHARE3 capabilities have been assessed both by already available experimental data supplied by ENEA (HERO-2 and PERSEO) and by new experimental campaigns carried out with PKL (FRAMATOME) and PASI (LUT University) facilities. A SACO system has been also connected to the secondary side of one steam generator of the 4-loop PWR integral test facility PKL, providing data to characterize the capability of the passive system to remove the decay heat in several operating conditions [6]. Within the framework of the H2020 ELSMOR project, post-test calculations were performed to demonstrate the capability of the RELAP code to simulate the phenomena occurring in a prototypical heat exchanger [4].

As part of broader international collaborations, ENEA coordinated the PERSEO benchmark [2] within the OECD/NEA WGAMA framework. This initiative aimed to evaluate the performance of computational tools in simulating passive safety functions, natural circulation stability, and thermal-hydraulic coupling between the primary system and containment. The PERSEO benchmark was specifically developed as part of the OECD/NEA's broader initiative on the reliability of thermal-hydraulic passive systems and contributed directly to the preparation of the Status Report on the Reliability of Thermal-Hydraulic Passive Systems [2]. In parallel, ENEA also participated in the OECD/NEA State-of-the-Art Report on Scaling in System Thermal-Hydraulics [7], which addresses the application of scaling methodologies to reactor safety and experimental design. This work is particularly relevant for the correct interpretation and extrapolation of results from IETF and SETF, including PERSEO, to full-scale reactor systems. For SA conditions, ASTEC (study carried out with ASTEC V2, IRSN all rights reserved, [2022]) and MELCOR were validated using QUENCH-6 [8, 9] tests and the integral PHEBUS facility [10], providing critical insights into fission product behaviour and core degradation, and uncertainty estimation. These activities have been done along the CRP I31033 on Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors [8] and the H2020 MUSA project [10].

Contribution Full plant analyses

In recent years, several full-plant application activities have been conducted within the NUC-ENER-SIC laboratory, focusing on key topics of current relevance in the application of DSA to nuclear reactor systems. Among these, particular attention has been given to the characterization of state-of-the-art computational codes for simulating phenomena relevant to SMRs, in support of the emerging European SMR licensing process. One significant study involves the simulation of DBA and BDBA scenarios in a generic iPWR, using the ASTEC (study carried out with ASTEC V2, IRSN all rights reserved, [2024]) and MELCOR codes and the related uncertainties [11-13]. This application contributed to assess the capability of these tools to model thermal-hydraulic behaviour under nominal conditions and to predict the evolution of SA phenomena under degraded boundary and initial conditions. From a thermal-hydraulic perspective, the analysis is complemented by simulations performed with the TRACE code. In addition, a dedicated study is currently in progress to investigate the feasibility of IVMR as a SA management strategy for SMRs. All these activities have been developed within the framework of the SASPAM-SA project [11] coordinated by ENEA. Activities are currently in progress on the applicability of the REPAS methodology to SMRs, using input and experience gained from the SASPAM-SA project [14]. These activities will be further developed within the EASI-SMR project, where the methodology will be applied to a generic European SMR design [15].

Conclusions

Ensuring the safety of nuclear technologies remains the fundamental prerequisite for their deployment and societal acceptance. The activities carried out within the NUC-ENER-SIC Laboratory over the past decade reflect a systematic and multi-faceted approach to supporting the safety demonstration of both current and advanced reactor designs, with a particular focus on SMRs. Through the integration of experimental campaigns (PERSEO, HERO, ELSMOR), rigorous scaling analyses, and the validation of thermal-hydraulic and severe accident codes (e.g., RELAP, TRACE, ASTEC, MELCOR), the laboratory has contributed to advancing the understanding of key phenomena relevant to passive safety systems. In particular, the use of methodologies such as BEPU and REPAS has strengthened the technical robustness of deterministic safety analyses, ensuring the traceability of results and the transparency of assumptions. The coordination of international benchmarks (e.g., PERSEO/OECD-NEA) has further emphasized the central role of experimental validation in qualifying computational tools and assessing their predictive capabilities under design and beyond-design conditions. Moreover, the full-plant simulations carried out in the framework of European initiatives like SASPAM-SA have provided a solid foundation for evaluating the feasibility of innovative accident management strategies, such as In-Vessel Melt Retention (IVMR), and for assessing the integration of passive systems in compact reactor configurations. These efforts contribute directly to the safe deployment of European SMRs and to the broader goal of achieving a net-zero carbon energy mix by 2050.

Acknowledgment

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INTEGRAL EFFECT PRELIMINARY TESTS IN CIRCE-THETIS FACILITY: HEAT LOSSES CHARACTERIZATION AND TRANSITION FROM FORCED TO NATURAL CIRCULATION EXPERIMENTS

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Abstract

In the framework of ENEA's research program focused on the development of Generation IV LFR (Lead-cooled Fast nuclear Reactor) technology, the CIRCE-THETIS experimental campaign holds a prominent position in the study of thermal-hydraulics of lead-cooled systems. The CIRCE facility, configured with the THETIS (Thermal-hydraulic HElical Tubes Innovative System) test section, enables the experimental reproduction of the thermal-hydraulic behavior of a lead-bismuth pool-type nuclear reactor. The test section comprises the key components of the primary circuit of a pool-type LFR: the electrical core simulator, the heat exchanger, and the mechanical pump. CIRCE is also equipped with a secondary loop, operating with demineralized water, and a steam line. Additionally, the THETIS campaign includes the investigation of the Reactor Vessel Auxiliary Cooling System (RVACS). During the pre-test phase, the thermal-hydraulic behavior of both the primary and secondary circuits was characterized, along with the assessment of the system's heat losses at various pool temperatures. The heat losses were identified as a thermal sink, which was necessary and sufficient to initiate natural circulation. Consequently, the transition from forced (FC) to natural circulation (NC) was studied following pump trips at different preset electrical core power levels. Measurements were then conducted to determine the mass flow rate through the core, the peak cladding temperature, and the pressure losses along the flow path of the primary system, with particular attention to those across the pump impeller.

Keywords: *Gen-IV; Heavy Liquid Metal; IET Facility; LFR; Natural Circulation.*

Introduction

The objective of this work is to present some preliminary results from the ongoing experimental campaign known as CIRCE-THETIS. The campaign is being conducted using the CIRCE experimental facility, located at the ENEA Brasimone Research Centre.

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The CIRCE-THETIS tests pursue multiple objectives: demonstrating the operation of the helical coil steam generator (HCSG) as a decay heat removal (DHR) system; demonstrating the functionality of the RVACS (Reactor Vessel Air Cooling System) as an alternative DHR system; and investigating the transition from HCSG-based to RVACS-based DHR.
Additional goals include the thermal-hydraulic characterization of the two prototypical primary system components: the steam generator and the main circulation pump (MCP). Finally, the campaign includes the study and characterization of part of the measurement instrumentation implemented in the primary system - specifically, the probes used to monitor oxygen content in the lead-bismuth eutectic and the so-called bubble tubes (BTs) sensors responsible for measuring pressure at various locations immersed in the primary pool.

CIRCE facility description

The CIRCE experimental facility [1] has been designed to operate with liquid lead-bismuth eutectic alloy (LBE), to study thermal-hydraulic phenomena with integral effects that characterize the operation of liquid-metal-cooled nuclear reactors. The facility consists of three tanks, Fig. 1: the main tank (S100), the storage tank (S200), and the transfer tank (S300). These tanks are interconnected by a piping system equipped with both manually operated and remotely actuated valves, specifically designed to function with liquid metals. All surfaces in contact with the liquid metal, whether for flow or stationary conditions, are heated and thermally insulated. Additionally, these surfaces are instrumented with a sufficient number of thermocouples to monitor the temperature distribution within the fluid domain. Inside the tanks, an inert cover gas region is maintained, monitored via dedicated pressure transducers and thermocouples. To track fluid levels, the tanks are also equipped with conductive point level probes, which have been designed, and manufactured at the ENEA Brasimone Research Center laboratories. Furthermore, the tanks are fitted with safety vent valves to manage potential overpressure conditions. The main CIRCE tank is designed to accommodate various test sections, depending on the specific experiment to be conducted. CIRCE is also suitable for testing full-scale prototype components, such as pumps and heat exchangers. For heat exchanger testing, the facility includes a secondary loop designed to operate with water as the working fluid. This loop comprises a pump (the FW pump), an electric heater, a steam line, and a bypass for startup transients. The secondary system is engineered to operate at pressures up to 200 bar. It is fully instrumented with thermocouples, pressure transducers, and flow meters to measure the relevant thermal-hydraulic parameters along the entire fluid path. The main tank is also capable of housing a core simulator, which is an electrically heated core mock-up designed to deliver up to 925 kW of electrical power. The core simulator consists of a hexagonal-section assembly containing 37 rods, each with a diameter of 8.2 mm and an active length of 1 m. The entire facility is fully remotely controllable, as it is equipped with an advanced control and data acquisition system.

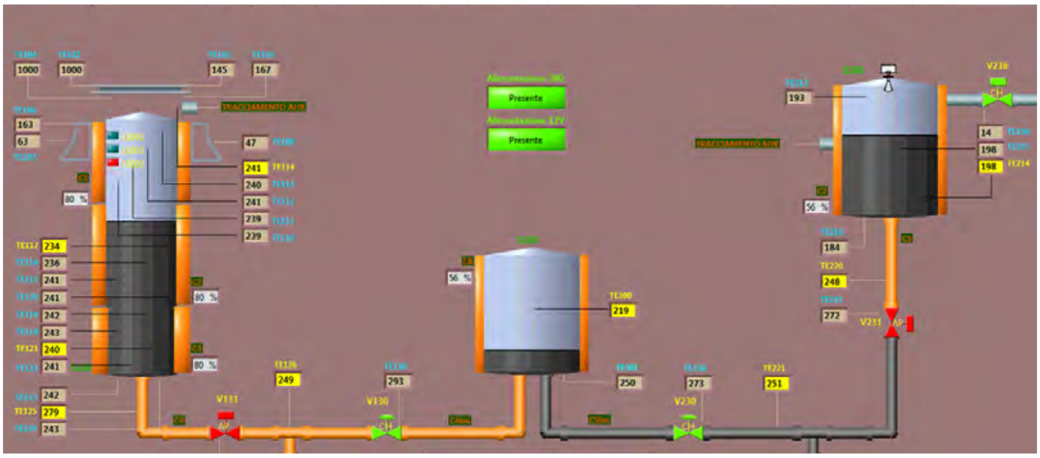

Fig. 1: CIRCE facility layout (SCADA view): main vessel, S100 (left); transfer tank, S300 (center); storage tank, S200 (right).

THETIS Test Section (TS) description

The THETIS [3] test section (Fig. 2, Fig. 3), consists of the prototypical helical coil steam generator (HCSG) [2], the main circulation pump (MCP), and the Fuel Pin Simulator (FPS). The FPS features an internal structure designed to isolate the electrical power supply cables from the LBE using a dedicated dead volume. To establish the flow path and the separation between the two pools – the hot and cold pools, characteristic of a pool-type LFR design – the internals include the fitting volume (FV), the riser pipe connecting the hot pool (Separator) to the FPS, and the can wall of the MCP.

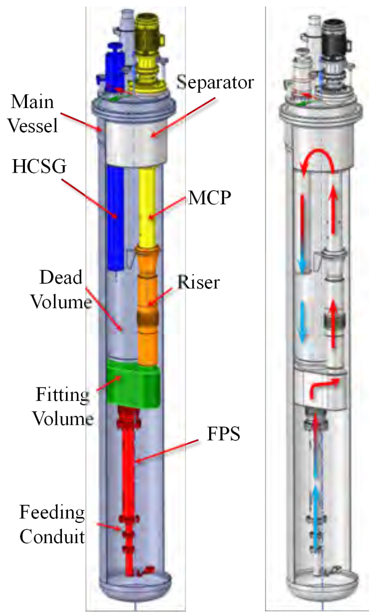

Fig. 2: CIRCE-THETIS Test Section [3]: schematic view oof the key components (left) and primary system flow path (right).

Fig. 3: CIRCE-THETIS Test Section.

The instrumentation used to measure the thermal-hydraulic parameters of the primary system includes thermocouples, pressure sensors, oxygen probes, level sensors, and a Venturi flow meter. The temperature field within the primary system is monitored by over 250 thermocouples, organized as follows:

- Pool thermocouples, distributed at various depths to monitor thermal stratification;
- Thermocouples for the HCSG, positioned to monitor the tube bundles and the shell-side flow channel;
- Thermocouples for the MCP, including locations at the suction and discharge, the can wall, and the MCP's cover gas;
- Cover gas thermocouples;
- Thermocouples for the FPS, specifically monitoring the flow channel;
- Thermocouples along the entire primary flow path: FV, riser, separator.

Outside the primary circuit - namely, outside the main vessel - wall thermocouples are installed on both the vessel and the thermal insulation layer to measure the system's thermal losses. In this regard, dedicated thermocouples monitor the forced cooling of the FPS power supply cables, coupled with an anemometer to evaluate the exchanged power during the process. Additionally, thermocouples are installed for the RVACS system.

Regarding the secondary system, the instrumentation includes thermocouples, pressure sensors, and flow meters.

Transition from FC to NC experiments

Before proceeding with the study of the system's behavior under natural circulation conditions, the overall thermal losses of the primary system were determined (Table 1). To quantify the heat losses of the entire primary system, thermal balance calculations were performed. Following this, the FPS system was regulated to supply the power level derived from these calculations. This power input was then fine-tuned to achieve a steady-state temperature within the S100 pool. During this phase, the MCP operated at a regime sufficient to ensure the nominal flow rate, equal to 35.7 kg/s through the FPS.

Table 1: CIRCE-THETIS Heat Losses preliminary characterization.

Steady State	**FPS [kW]**	**MCP [kg/s]**	**Pool Temp. [°C]**	**ΔT_{FPS}[°C]**
#1	25	35.7	410	5
#2	50	35.7	450	9.5

Once a steady-state temperature field was established in the pool, a pump trip was initiated. The ensuing transient leading to the onset of NC was then observed. The system subsequently reached a new steady state under NC conditions. As shown in Fig. 4, the system had reached steady-state conditions with the FPS operating at 50 kW, effectively balancing system's thermal losses and maintaining a primary pool temperature of 450°C (T-MS-080 in Fig. 4). Immediately following the MCP trip (Fig. 5:), a temperature peak (T_{clad} in Fig. 4) was recorded on the clad of the pin in the hot channel, reaching 540°C. It is worth highlighting that the cladding temperature T_{clad} is a calculated parameter, considering the Ushakov correlation for the calculation of the heat transfer coefficient. Finally, the trend of the mass flow rate clearly indicates the onset of natural circulation, stabilizing at approximately 10 kg/s within about 3.5 minutes (Fig. 4, Fig. 5).

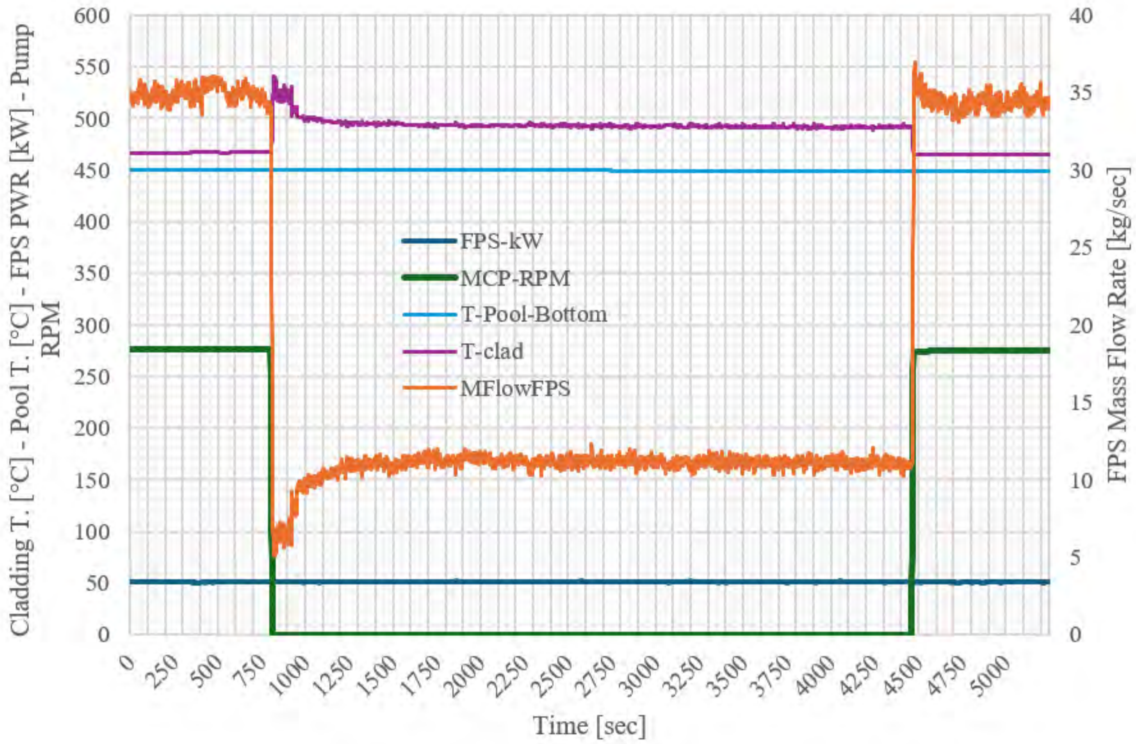

Fig. 4: Transition from FC to NC in CIRCE-THETIS: time trends of the key parameters.

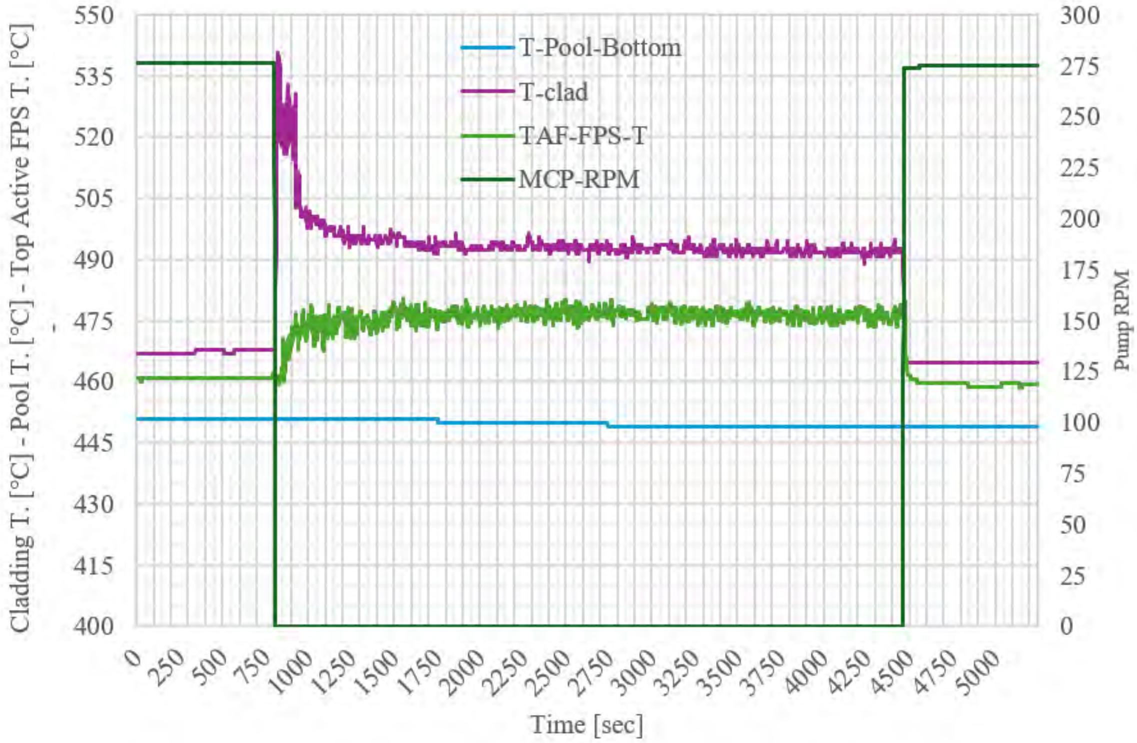

Fig. 5: Cladding temperature VS Main Coolant Pump RPM.

Conclusions

The system exhibited a strong tendency to establish natural circulation under all imposed core power conditions, relying solely on the thermal sink effect provided by the system's heat dispersion. The data acquisition and measurement system proved efficient and reliable in capturing all relevant phenomena and thermal-hydraulic quantities, particularly the mass flow rate through the core under various natural circulation regimes, the peak temperature on the pin in the FPS' hot channel, the coolant pressure losses, the level differences between the hot and cold pools, and the temperature stratification within the pool.

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APPLICATION OF THE REPAS METHODOLOGY TO ASSESS THE RELIABILITY OF THE EHRS IN A GENERIC iPWR SMR

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Abstract

Small Modular Reactors (SMRs), in particular Light Water-SMR, represent a promising advancement in nuclear technology, offering enhanced safety through the integration of passive safety systems. However, the reliability of these passive systems, particularly concerning Natural Circulation (NC) mechanisms, remains a critical issue. The Emergency Heat Removal System (EHRS) is a key passive system designed specifically to manage decay heat removal during accident conditions. This work presents a first attempt to apply the REPAS methodology to assess the performance of the EHRS in a generic integral-Pressurized Water Reactor (iPWR) SMR. The methodology combines probabilistic and deterministic approaches to evaluate failure probabilities under various scenarios, including a Design Basis Accident (DBA) involving a double-ended guillotine break of a Direct Vessel Injection (DVI) line. MELCOR, a severe accident (SA) simulation code, is used to develop the model and perform the calculations. The analysis identifies key failure conditions, such as low valve opening ratios, high concentrations of non-condensable gases, and low Refueling Water Storage Tank (RWST) levels, which could compromise system functionality. The results also highlight the importance of conducting a thorough uncertainty analysis and carefully selecting the probability density functions (PDFs) within the REPAS methodology to ensure a reliable assessment of the passive safety systems' performance and potential functional failures.

Keywords: *REPAS; EHRS; Passive Systems; SMR; DBA*

Introduction

Ensuring the safety of Nuclear Power Plants is fundamental to the continued development and acceptance of nuclear energy. Light Water-SMRs, and in particular integral-Pressurized Water Reactor (iPWRs), represent a promising evolution of current designs, emphasizing passive safety features that reduce reliance on external power and active systems.

While passive systems enhance inherent safety, their effectiveness depends on physical phenomena like Natural Circulation (NC), which are sensitive to scenario-specific conditions and operational uncertainties. Thus, there is a risk of functional failure, i.e. failure of passive systems to operate as

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expected due to limitations in natural driving forces, even in the absence of any mechanical or electrical malfunction. To analyze this failure, the REPAS methodology [1] [2] combines deterministic and probabilistic approaches to evaluate the reliability of passive safety systems under accident scenarios. This study applies REPAS to a generic iPWR design, focusing on the EHRS and using MELCOR [3] code for transient simulations.

REPAS methodology

The Reliability Evaluation of Passive Safety Systems (REPAS) is a methodology based on the evaluation of the failure probability of a passive system to perform its desired function, considering the uncertainties of the physical and geometric parameters that can cause the failure of the expected thermal-hydraulic phenomena and, consequently, the passive safety system. The general objective of REPAS is to analytically characterize the performance of a passive system to ensure its correct operation and to compare the performance of passive and active systems. The key points of the adopted methodology are listed below in order:

1. Characterization of the system design, operating conditions and Target Mission (TM).
2. Identification of design and critical parameters, with their reference values, ranges of variability and probability distributions.
3. Code modeling of the system and execution of the reference calculation.
4. Identification of the system Failure Criteria (FC).
5. Execution of repeated code runs.
6. Quantitative reliability estimation.

Reference Design and Calculation

The REPAS methodology is applied to a generic 300 MWe SMR of the iPWR type, known as Design 2, developed within the Euratom SASPAM-SA project (Figure 1) [3]. This design includes a dry containment and an integral Reactor Pressure Vessel (RPV) that houses the core, steam generator, control rod drive mechanism, pressurizer, and primary pumps. Design 2 also presents several passive safety systems: Emergency Heat Removal System (EHRS), Automatic Depressurization System (ADS), Emergency Borated Tank (EBT), Pressure Suppression System (PSS), Long-term Gravity Make-up System (LGMS), and Reactor Cavity flooding.

The SA integral code MELCOR was used for deterministic simulations. The input-deck was built with a nodalization based on a generic IRIS-type SMR using public data [5] and scaled parameters from the SPES-3 facility [6], as shown in Figure 2.

The selected transient is a Loss Of Coolant Accident (LOCA), specifically a double-ended guillotine break of one of the two DVI lines, with all passive systems available [6].

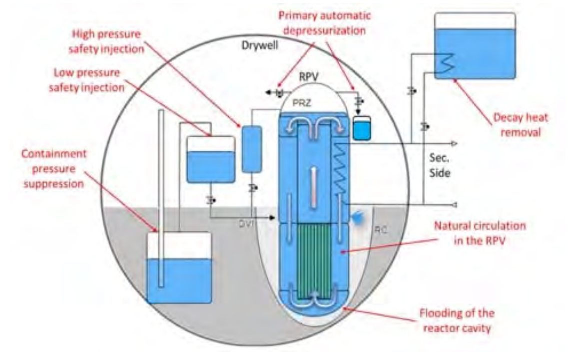

Fig. 1: Generic Design 2 layout.

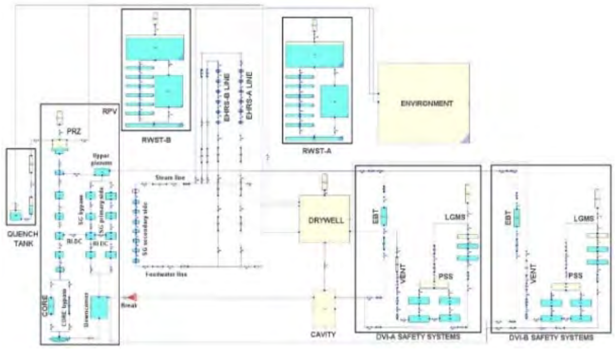

Fig. 2: MELCOR nodalization.

Application of the REPAS

The REPAS methodology is applied to evaluate the EHRS and its impact on the long-term core cooling. Three FC were defined based on the reference calculation values:

- **FC1**: Maximum cladding temperature > 600 K after EHRS activation (in reference calculation the final temperature is ≈ 400 K, as shown in Figure 3).
- **FC2**: Reactor Pressure Vessel (RPV) collapsed level < ½ active core height (in reference calculation it remains above Top of Active Fuel, as shown in Figure 4).
- **FC3**: EHRS removed power < 70 MW at activation of the system (in reference calculation is > 100 MW, as shown in Figure 5).

The TM under analysis is the removal of decay heat from the core to an external sink. Five parameters influencing EHRS performance were selected (Table 1): loss coefficient, surface roughness, valve opening ratio (VOR), non-condensable gas (NCG) fraction, and pool initial level. Each parameter was assigned a discrete probability distribution, and from the nominal values of the reference case, considering physical-technical constraints driving the parameter limits, ranges of variability were assigned.

Table 2: Selected parameters for the REPAS probabilistic sampling.

Parameter	Nominal value	Range	PDF
Loss coefficient	0.6	+100%	Piecewise
Surface roughness	5.0e-5 m	±0.05%	Normal
Valve opening ratio (VOR)	1	-100%	Piecewise
EHRS-HX initial level (related to NCG)	0.1667 m	-100%	Piecewise
Pool initial level	2 m	-37.5%	Piecewise

Due to input-deck constraints, the NCG fraction was sampled indirectly by varying the initial liquid level in the EHRS heat exchanger, as this is inversely related to the presence of NCG.

Results

A total of 150 runs were performed, and in each run the selected parameters were randomly sampled. Of these simulations only 126 were analysed because of 24 numerical failures (still more than the 124 required by Wilks' formula for the study to exhibit good statistical reliability). A deterministic simulation was also performed to evaluate a low probability extreme case not captured by probabilistic sampling. The deterministic calculation involved the case of VOR equal to zero, which is an extreme within the range of parameter variation. The latter simulation results in a SA due to the inability to cool the uncovered core.

Regarding the probability calculations, Figures 3, 4 and 5 present the results respect to cladding temperature, RPV level, and EHRS power removed.

We observe that: FC1 was exceeded in seven probabilistic runs (Figure 3), with six showing significant exceedance by the end of the simulation, FC2 was exceeded in five runs (Figure 4), and FC3 was exceeded in twenty-three runs (Figure 5), with sixteen showing only this criterion being violated, while the remaining cases overlap with previously mentioned exceedances.

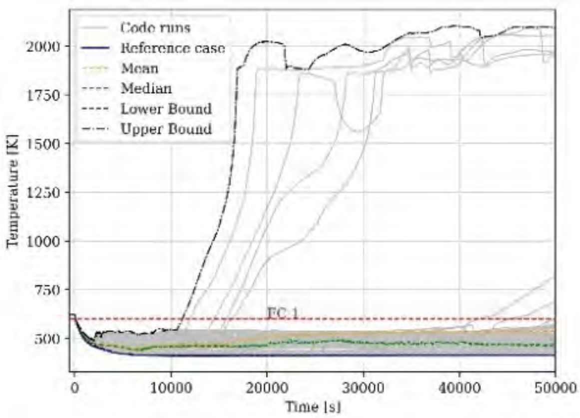

Fig. 3: Cladding temperature in probabilistic calculations.

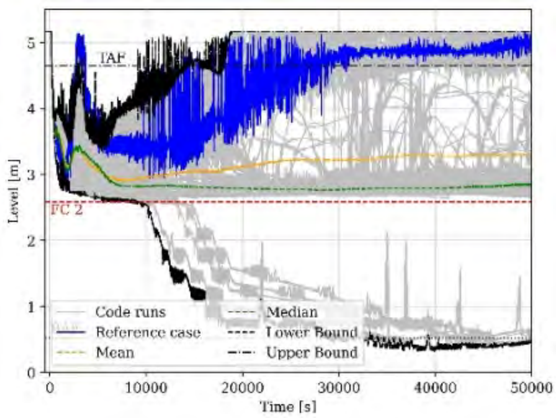

Fig. 4: RPV collapsed coolant level in probabilistic calculations.

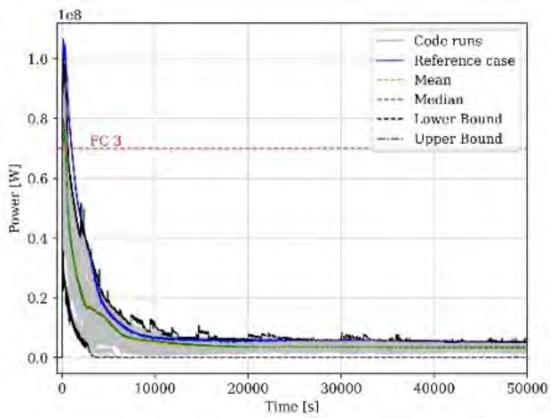

Fig. 5: Power removed by EHRS in probabilistic calculations.

In the following Figure 6 the maximum cladding temperature is plotted against its probability of occurrence for each simulation, with the deterministic calculation result marked by a black dot. Note that some simulations despite the low probability of occurrence still met TM. This is given by the loss coefficient and surface roughness values, which despite being sampled with extreme (and thus low probability) values compared to the nominal, still have limited influence on the system behaviour, so their impact on reliability is minimal.

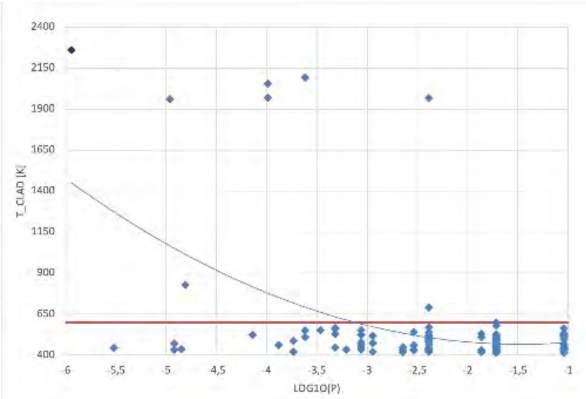

Fig. 6: Maximum cladding temperature as a function of the occurrence probability.

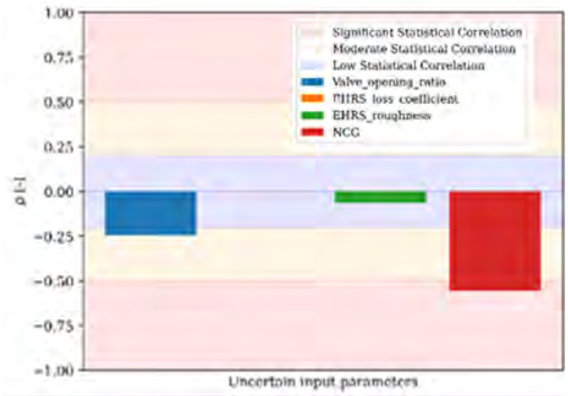

Fig. 7: Spearman's correlation coefficient for maximum cladding temperature.

To identify the most influential parameters, Spearman correlation coefficients were calculated and in Figure 7, those related to cladding temperature can be observed. The analysis shows that the NCG fraction (which we recall should be interpreted inversely because of the inverse correlation between NCG and liquid level of EHRS heat exchanger; see final part of section "Application of the REPAS") negatively affects cooling performance, while the VOR enhances power removal and reduces cladding temperature. In contrast, the EHRS loss coefficient and surface roughness have negligible effects.

Conclusions

This study applied the REPAS methodology to a generic 300 MWe iPWR, analyzing the EHRS system in a scenario of guillotine rupture of a DVI line. Using MELCOR, 150 probabilistic simulations and one deterministic case (with full EHRS failure, VOR = 0%) were performed. Results confirm the applicability of REPAS to EHRS assessment and highlight the need for targeted deterministic runs to capture low-probability or limit conditions. It was observed that functional failures are mainly related to very low values of VOR, high concentrations of NCG, and low levels of RWST. Uncertainty analysis shows that reliability results are very sensitive to the definition of input uncertainties. Therefore, the selection of probability distributions and parameter ranges must be based on sound references, combining experimental data, analysis, and expert judgment. Future work should refine these inputs to obtain more robust assessments.

Acknowledgment

ENEA carried out the research activity with MELCOR code and SNAP, obtained in the framework of the ENEA-ISIN agreement signed on March 23rd 2020 as part of the General Arrangement between the United States Nuclear Regulatory Commission (US-NRC) and the Italian National Inspectorate for Nuclear Safety and Radiation Protection (ISIN).

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SUPPORTING THE ESFR-SMR DESIGN: SAFETY ANALYSES AND FUEL OPTIMIZATION ACTIVITIES AT ENEA

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Abstract

The Small Modular Reactors (SMRs) are gaining increasing attention as a promising solution to complement large-scale nuclear reactors in future low-carbon energy systems. Thanks to their reduced size, modular design, and enhanced safety features, SMRs offer flexible integration into the grid, economic advantages through standardization and production in series, and flexibility. Among the advanced SMR concepts, sodium-cooled fast reactors (SFRs) are interesting for their potential in closing the fuel cycle.

In this framework, the ESFR-SIMPLE project, funded by the Euratom research programme, aims to redefine the European-SFR (ESFR) concept by exploring innovative technologies that enhance safety, economic performance, and public acceptance. The key goal of the project is the conceptual development of an SMR version derived from the full scale ESFR (3600 MWt), focusing on simplification, use of passive systems, and improved component integration.

ENEA contributes to this effort through activities in "SMR safety studies" and "Optimised fuel element" work packages. This paper briefly collects the status of these activities that will support the design choices for the ESFR-SMR, aiming to improve the safety margins and fuel performance.

Keywords: *ESFR-SIMPLE; RELAP5; SFR; SMR; TRANSURANUS*

Introduction

Sodium-cooled fast reactors are among the most advanced Generation IV technologies, benefiting from over sixty years of operational experience. They offer high neutron economy, efficient fuel use, and potential for closing the fuel cycle with plutonium and depleted uranium, key features for long-term sustainability and waste reduction.

In Europe, SFR development has progressed through collaborative projects. CP-ESFR (2009–2012) produced the first 3600 MWth pool-type design, while ESFR-SMART (2017–2022) updated it with safety enhancements such as a heterogeneous core, improved decay heat removal by natural convection, and structural reinforcements aligned with post-Fukushima requirements.

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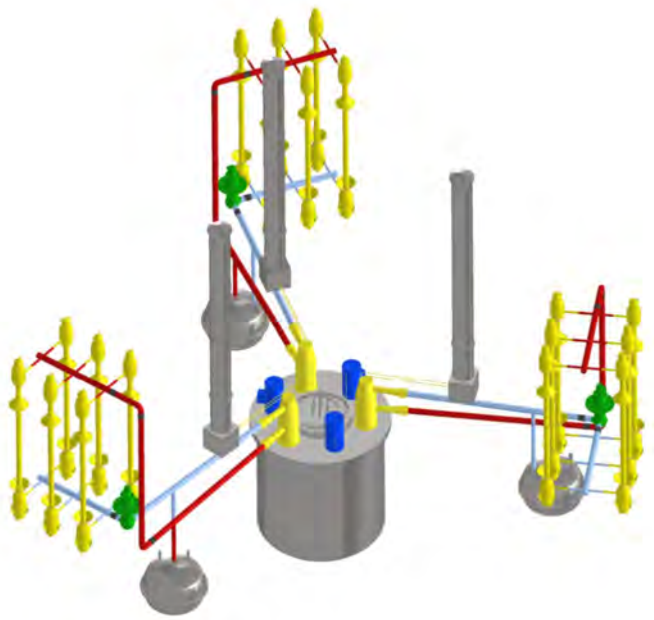
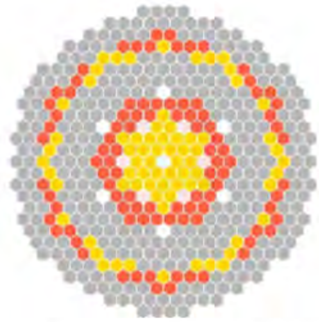
Launched in 2022, the ESFR-SIMPLE project aims to improve SFR safety and economics through innovative design of the ESFR-SMR, a 360 MWth Small Modular Reactor version of the original reactor, relying on passive safety systems.
Within this framework, ENEA contributes to two key work packages, described in the following chapters.

SMR safety studies (WP4)

The activity of WP4 is focused on the assessment of the ESFR-SMR response to protected and unprotected transients (e.g., PLOF, PSBO, ULOF, UTOP) in particular on decay heat removal capabilities under natural circulation conditions. The activity is currently at a preliminary stage, as it depends on data from other work packages related to the core [1] and system design [2] of the SMR reactor.
The ESFR-SMR consists of a pool-type primary circuit connected to three sodium-cooled secondary loops via intermediate heat exchangers (IHX). Each secondary loop transfers heat to tertiary water-cooled circuits through six steam generators, for a total of 18 SGs. Three centrifugal primary pumps circulate sodium from the cold pool through the diagrid into the core, with a small flow diverted to the vessel cooling system to protect the vessel wall.
The proposed core, shown in Fig.1, has reduced dimensions in both height and radius respect the full-size reactor and incorporates design adaptations such as higher plutonium content and the inclusion of a fertile plate in the inner core to mitigate positive reactivity insertion during sodium boiling events. Despite the downsizing, the design retains features like: 6 corium discharge tubes, 6 Control/Shutdown Devices (CSD) plus a Diverse Shutdown Device (DSD), a common fuel configuration for inner and outer zones, and a multi-ring reflector system. This configuration aims to ensure negative global void reactivity and improved safety behaviour in accidental transients like ULOF.
The secondary circuits have been simplified. In SMR version, the piping between IHXs and SGs is straight, and the thermal expansion compensated with bellows. Each loop is provided with a centrifugal pump.
In case of reactor SCRAM, the DHRs ensure passive decay heat removal. Their design is still under development but is expected to follow the approach defined for the full-scale reactor.
The simulations will be performed with RELAP5mod3.3 [3] best-estimate system thermal-hydraulic code. Developed for the analysis of transients and accidents in water-cooled reactors, its capability has been extended by ENEA implementing the sodium fluid thermodynamic characteristics.
The first results obtained are related to the core static benchmark. Fig. 2 shows the temperature profiles of fuel, clad and coolant, both for inner and outer core hot channels.





S/A Type	# of S/A
Inner Fuel	36
Outer Fuel	48
CSD/DSD	6/1
Reflector (1st ring)	42/48/54
Spent Inner	36
Spent Outer	48
Corium discharge	6
Reflector (2nd ring)	150

Fig.1: ESFR-SMR scheme and details of the core sub-assembly (S/A) distribution.

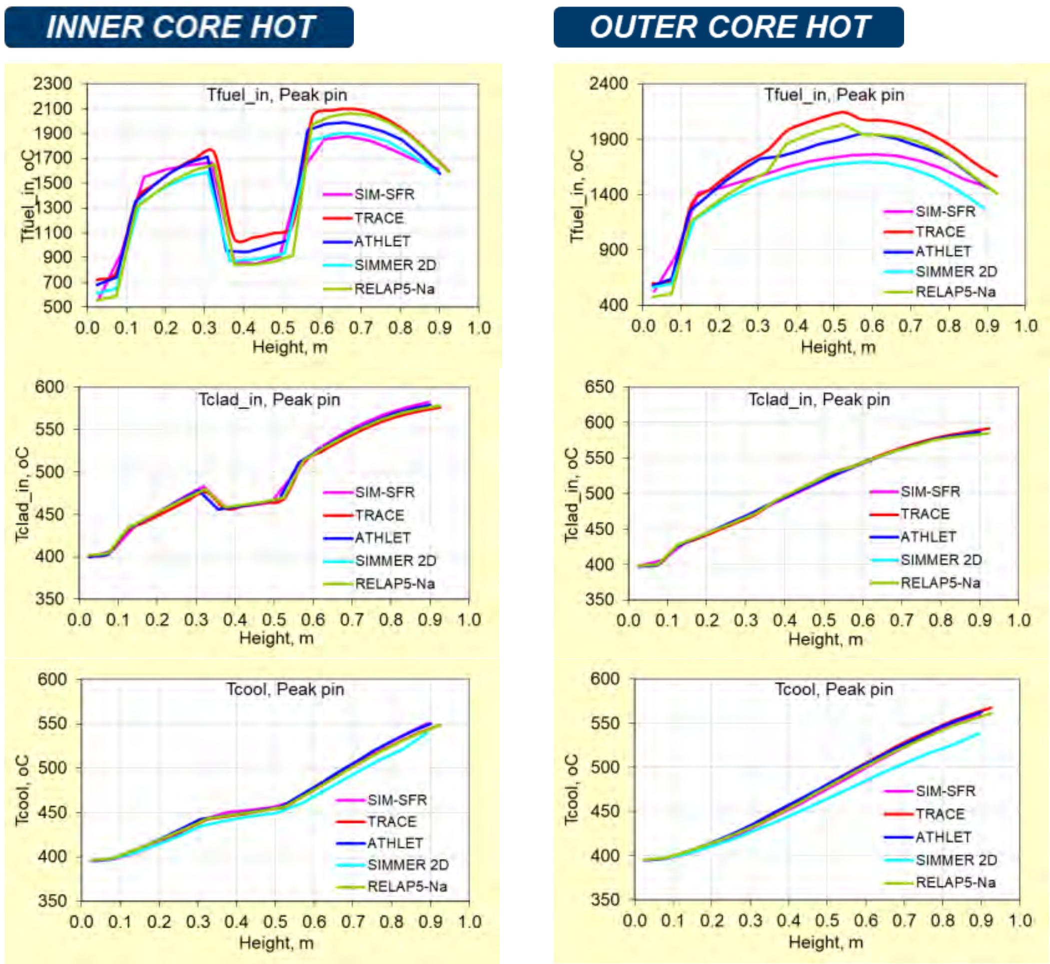


Fig.2: Main results of the preliminary core static benchmark.

Optimization of MOX fuel element

As part of WP8, a benchmark exercise was conducted to assess the transient behaviour of MOX fuel pins using several fuel performance codes by five participating institutions. Two transient scenarios are considered: Unprotected Loss of Flow (ULOF) and Unprotected Transient Overpower (UTOP). Three MOX pin designs derived from previous experience with SFRs were assessed under these conditions. The reference design is based on the ESFR fuel pin, and the other two designs being the Superphénix (SPX) and CAPRA configurations.

Table 1: ESFR-SIMPLE parametric designs geometric specifications (mm).

	Parameter	**ESFR**	**SPX**	**CAPRA**
Cladding	Clad outer diameter	10,67	8,45	6,73
	Clad inner diameter	9,67	7,45	5,73
Upper plenum	Length	50	50	50
Fissile fuel	Pellet outer diameter	9,36	7,14	5,42
	Pellet inner diameter	3,12	2	2
	Column length	750	750	750
Fertile fuel	Outer diameter	9,36	7,14	5,42
	Inner diameter	0,00	0,00	0,00
	Length	250	250	250
Lower plenum	Length	913	913	913

All the considered designs are modelled with Oxide dispersion-strengthened (ODS) steels cladding and the primary geometric specifications are provided in Table 1.

TRANSURANUS, developed by the European Commission's JRC, is a code for analysing the thermal/mechanical behaviour of nuclear fuel rods under various conditions. For this benchmark, version *v1m1j24* was used with models adapted for high-Pu MOX fuel, including:

- A mechanistic model for fission gas release and swelling, linked to hydrostatic stress [4].
- An LHR-dependent model for fuel cracking [5].
- A gap conductivity model based on ESFR-SMART data [6].

Model choices and material properties reflect the current state-of-the-art and prior benchmarking experience with MOX fuels under transient conditions.

Only TRANSURANUS results are presented. Participants first performed steady-state simulations to set initial conditions, followed by transients varying parameters like reactivity insertion and coolant flow degradation. The goal was to assess fuel melting margins and cladding failure risks. The benchmark allowed code-to-code comparison and revealed design-specific behaviours and sensitivities to input assumptions.

The simulations assumed two-cycle irradiation under ESFR conditions. Steady-state results showed wide burnup variation at same Linear Heat Rate (LHR), with CAPRA the highest and ESFR the lowest, mainly due to fuel diameter, affecting pre-conditioning before the transient.

In the first transient (UTOP), reactivity insertions of 280 and 400 pcm were analysed. Fuel melting occurred in CAPRA and SPX, while ESFR only at 400 pcm (Fig. 3). Geometry played a key role, since the ESFR lower volumetric power density limited temperature rise, reducing melting risk. CAPRA exhibited the highest melted fraction and centerline temperatures comparable to SPX, owing to its more compact geometry.

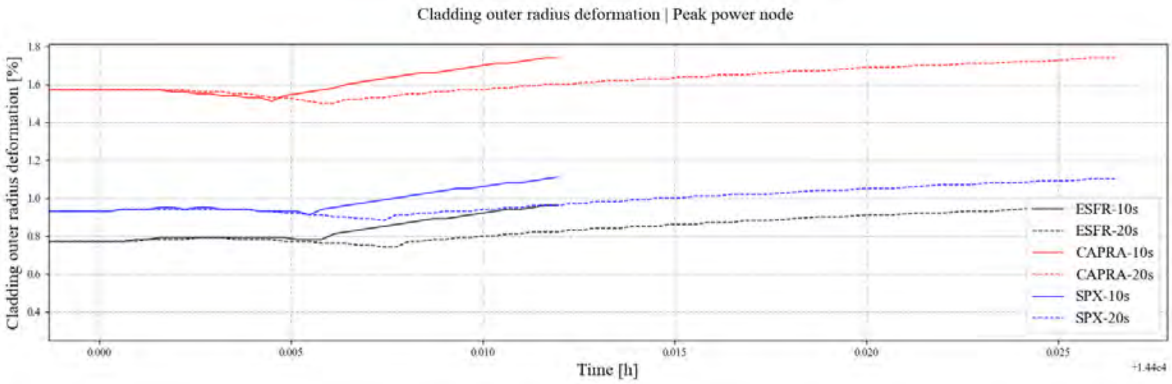


Fig.3: ESFR-SIMPLE Fuel melting fraction after UTOP transient.

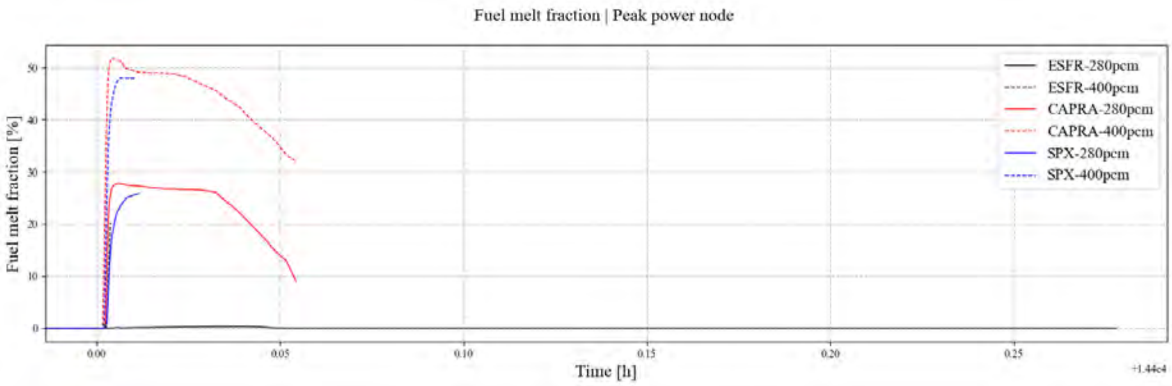


Fig.4: ESFR-SIMPLE designs fuel cladding deformation after ULOF transient.

In the second transient (ULOF), the loss of primary coolant flow due to pump failure without safety intervention was simulated over 10 and 20 seconds, leading to natural circulation. Although LHR decreased, cladding temperatures rose steadily, compromising the clad integrity due to high temperature and rising internal gas pressure, causing significant deformation (Fig. 4).

Cladding behaviour was driven by the interaction of temperature, pressure, and fuel swelling. All pins showed similar trends, but CAPRA exhibited higher nominal values, likely due to greater burnup. A full assessment is limited by scarce data on ODS steel under transient, high-temperature conditions.

Conclusions

The ESFR-SIMPLE project aims to design a 360 MWth SMR reactor cooled by sodium. ENEA contributes through activities on SMR safety studies and optimization of fuel elements.

The activity carried out so far in WP4, focused on assessing the ESFR-SMR response to protected and unprotected transients, is still at a preliminary stage. The initial results obtained with the RELAP5 model under nominal steady-state core conditions are promising, but the best is yet to come. In the near future, the model will be enhanced to include point kinetics to account the neutronic feedbacks, a detailed thermal-hydraulic description of the primary, secondary, and tertiary circuits (the latter limited to SGs with boundary conditions), and the DHR systems, following the approach previously adopted in the ESFR-SMART project.

The benchmark carried out in WP8 on MOX fuel pin revealed clear differences in pin behaviour. The ESFR design showed the best thermal response due to its geometry but still melting under severe transients. CAPRA, with higher burnup, experienced earlier fuel melting and cladding stress. The study highlighted the impact of pellet geometry, cladding behaviour, and transient dynamics on MOX performance. Results underline the need for continued design refinement and further development of models in TRANSURANUS for improved accuracy in transient scenarios, including an expansion of the code database of ODS material properties.

Acknowledgment

The ESFR-SIMPLE project is funded by the European Union under Grant Agreement 101059543. Views and opinions expressed are however those of the authors only and do not necessarily reflect those of the European Union. The European Union cannot be held responsible for them.

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SESSION 2

INNOVATIVE DESIGN AND MODELLING OF NUCLEAR SYSTEMS

Chairperson: *P. Console Camprini*

Session 2, entitled "Innovative Design and Modelling of Nuclear Systems", was aimed at providing an overview of a set of activities addressing advanced methodologies and tools for the design, analysis, and qualification of innovative nuclear systems. The session focused on one of the key strategic lines of action of ENEA, namely the continuous development and integration of experimental research, modelling capabilities, and design-oriented analytical tools to support advanced reactor concepts, with particular emphasis on lead-cooled fast reactors and Small Modular Reactors. Most of the presented activities are framed within national and international research programmes and collaborative initiatives, and are aimed at strengthening the capabilities for predictive simulation, validation, and optimization of nuclear systems. The contributions highlighted the use of advanced thermal-hydraulic, multi-physics, and system-level modelling approaches, supported by dedicated experimental facilities and benchmark activities, to improve the accuracy and reliability of design and analysis methodologies.

Several presentations addressed the role of experimental infrastructures and coordinated international efforts in supporting code validation and model qualification. In this context, the outcomes of international benchmarks and coordinated research projects were illustrated, demonstrating the effectiveness of combining experimental data with numerical simulations based on system codes, CFD tools, sub-channel approaches, and coupled multi-scale methodologies. These activities contribute to consolidating a robust knowledge base for the analysis of complex phenomena, such as the transition from forced to natural circulation and the thermal-hydraulic behaviour of advanced reactor components.

Other contributions emphasized the importance of a holistic and design-oriented approach to reactor core development. The concept of a virtuous cycle linking design needs, experimental data, code development, validation, and application was presented as a key element to ensure continuous improvement of design capabilities. This approach enables the identification of priority areas for research and development, supports the optimization of reactor performance, and ensures the traceability and robustness of design choices throughout the development process.

Finally, the session highlighted the strategic role of high-quality nuclear data and advanced experimental programmes in enhancing the accuracy of reactor analyses. Activities focused on differential and integral measurements, sensitivity and uncertainty analyses, and targeted experimental campaigns were presented as essential enablers for reducing uncertainties in simulations and improving the representativeness of experimental facilities with respect to innovative reactor concepts.

Overall, the session demonstrated how the integration of advanced modelling tools, dedicated experimental research, and coordinated international activities constitutes a fundamental pillar for maintaining and strengthening ENEA's role in the innovative design and analysis of nuclear systems, providing qualified support to research, industry, and future reactor development programmes.

THE IAEA COORDINATED RESEARCH PROJECT ON NACIE-UP FACILITY: OUTCOME OF ENEA ACTIVITIES

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Abstract

The IAEA Coordinated Research Project on the NACIE-UP facility is an international benchmark focusing on thermal-hydraulic analysis of experimental tests performed at ENEA Brasimone. NACIE-UP is a loop facility whose main components are a Fuel Pin Bundle Simulator (FPBS), consisting of 19 wire-spaced and electrically heated pins, a heat exchanger and a gas lift pump. In 2017, experiments were performed to study the forced-to-natural convection transition under uniform and non-uniform power distributions in the FPBS, using liquid Lead Bismuth Eutectic metal as operative fluid. The goal of the benchmark is to assess the capabilities of numerical codes in simulating heavy liquid metal systems, using the experimental data, made available by ENEA, as reference for evaluation. The targeted codes are standalone system codes and coupled CFD-system codes, for the simulation of the whole loop facility; CFD and sub-channel codes, for the focused simulation of the FPBS. Moreover, to evaluate the effect of modelling assumptions for a complex system as NACIE-UP, the benchmark test cases concern both open and blind simulation phases, i.e., with and without reference data being a priori available to the benchmark participants.

ENEA, as a participant, has been involved in all the activities for code validation, using OpenFOAM and Ansys-CFX as CFD codes, RELAP5 as system code and the in-house developed ANTEO+ sub-channel code. Good results, in terms of deviations from the reference experimental data, were obtained and presented here, proving the adequacy of the methodologies and codes.

Keywords: *Code Validation; International Benchmark; Lead Bismuth Eutectic; NACIE-UP; Thermal-hydraulics*

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Introduction

The present paper provides a concise overview of the ENEA activities performed in the framework of the IAEA Coordinated Research Project on the NACIE-UP facility. The benchmark was proposed to assess the capabilities of numerical codes in simulating Heavy Liquid Metal (HLM) systems using experimental data collected from the NACIE-UP facility, located at ENEA Brasimone, as reference values [1]. ENEA was thus involved both as coordinator and as participant in the benchmark.

The NACIE-UP facility consists of a rectangular loop whose main components are a Fuel Pin Bundle Simulator (FPBS), i.e., a hexagonal bundle of wire wrapped pins that are electrically heated, a gas lift pump, and a heat exchanger. The loop is instrumented with several thermocouples (many of them are placed in the FPBS), pressure transducers and a thermal flow meter, as shown on the left of Fig. 1. The picture on the right of Fig. 1 shows the thermocouple locations (both on sub-channel bulk and pin walls) inside the FPBS; in particular, three cross section planes have been instrumented inside the FPBS. Several experiments were performed in 2017 to study the transition from forced to natural convection, following a shutdown of the gas lift pump. The NACIE-UP FPBS was designed to allow the possibility of having active pins (electrically heated) and non-active pins, to study different working setups of the FPBS.

The benchmark activities were divided into five different Work Packages: numerical simulations with system codes in WP1, with CFD codes in WP2, with sub-channel codes in WP3 and with coupled CFD-system codes in WP4. The last Work Package, WP5, focused on uncertainty quantification and was not related to a particular family of numerical codes.

The experimental cases that were chosen for the benchmark are: ADP10, where all 19 pins are active; ADP06, where only the inner 7 pins are active; and ADP07, where only a corner sector of 9 pins is active. Each case consists of a forced convection steady state, a transient condition following the gas lift pump shutdown, with the transition from forced to natural convection, and a final steady state with natural convection. While WP1 and WP4 focused on the simulation of the whole facility and experiment (steady states and transients), WP2 and WP3 only considered the initial and final steady state conditions of each experiment and only for the FPBS. In the following, some of the obtained results are shown for the initial steady state conditions of ADP 10 and ADP 07, which have been the simplest and the most challenging to simulate.

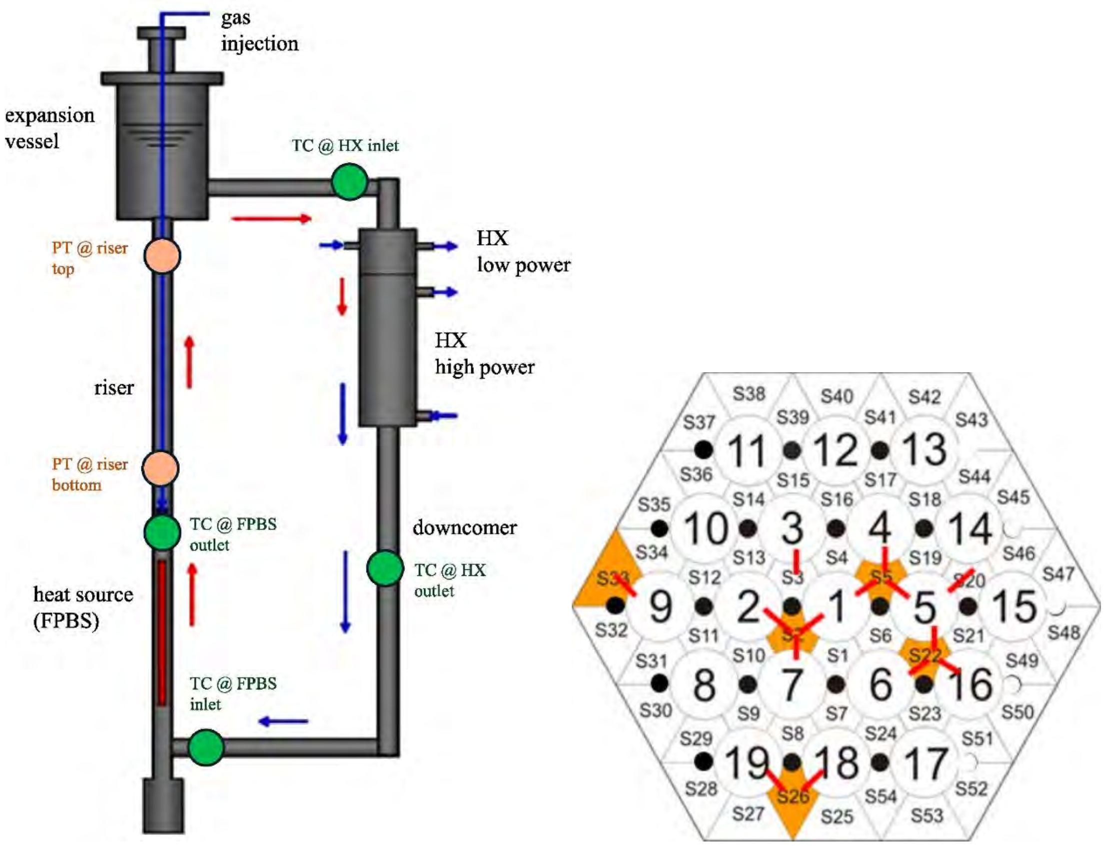

Fig.1: Sketch of the NACIE-UP facility (left) with highlighted positions of some of the thermocouples and pressure transducers [2] and highlighted thermocouple locations on the NACIE-UP bundle cross section (right) [1].

Methodology and Research Activity

FPBS results with CFD and sub-channel codes

The NACIE-UP FPBS was instrumented with approximately 60 thermocouples, placed on both sub-channel bulk positions and pin external surfaces as shown on the right of Fig. 1. The collected experimental data, for this component, allow to provide a detailed description of the system, making it very suitable for comparison with CFD and sub-channel codes. As regards the ENEA contribution, two CFD codes were used: OpenFOAM and ANSYS CFX, testing two different meshing approaches for discretizing the FBPS geometry. In both cases, the final computational models contained the fluid and solid pin domains, so to simulate the heat transfer through the pins, which was found to be a key feature in the cases where non-active pins were present. In active pins, a volumetric heat source was simulated to mimic the heat generation of the real pins. As sub-channel code, for WP3, the ANTEO+ code was used [3]. In this case, the interior of the pins was not simulated, so a uniform heat flux has been considered on the pin-fluid boundaries, with a model for simulating the azimuthal heat conduction on the pin external surface.

Table 1: Global evaluations of temperature differences [°C] between code computed and experimental values, on thermocouple locations for ADP 10 and ADP 07 initial steady state condition.

	ADP 10 SS1		**ADP 07 SS1**	
Case	Mean (abs)	Max (abs)	Mean (abs)	Max (abs)
OF	1.7	6.8	8.9	34.2
CFX	1.6	5.4	7.0	28.2
ANTEO+	2.8	8.9	13.7	48.2

A summary of the obtained results is reported in Table 1, for the ADP 10 and ADP 07 initial steady conditions, in the form of mean and maximum absolute temperature differences between code computed and experimental values. Very good agreement of results, with experimental data, is observed for the ADP 10 case, while for ADP 07, the found discrepancies are bigger. This was a more challenging case to simulate due to the non-symmetric power generation inside the bundle and the strong temperature gradients observed in the temperature field, as it can be appreciated in Fig. 2, where a comparison of the obtained temperature fields, from OpenFOAM, Ansys CFX and ANTEO+ codes is shown on the last of the FPBS monitored planes for ADP 07 case. Even though with a less detailed resolution method, the sub-channel averaged ANTEO+ solution matches well the temperature field computed by the much more detailed CFD codes. Since the computational model used in Ansys CFX was also used for the coupled CFD and system-code simulations, it also contained the wrapper solid domain, which provided an important thermal inertia contribution during the transients (after the shutdown of the gas lift pump).

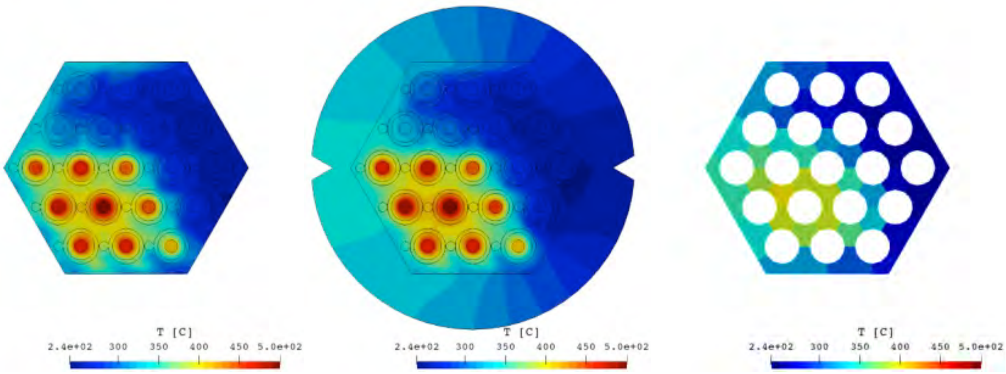

Fig.2: Cross section view of the simulated ADP 07 SS1 temperature field, on plane C, from OpenFOAM (left), Ansys CFX (middle) and ANTEO+ (right) results.

Whole facility simulations using system and coupled CFD-system codes

A model of the NACIE-UP facility has been realized through the RELAP5/Mod3.3 code, properly modified to implement the HLM thermo-physical properties. The sliced modeling approach has been adopted for the entire facility to better capture the behaviour of the system under natural circulation conditions. With the model pressure drops calibrated for the natural circulation regime, a simulation of only the first steady state has evidenced an underprediction of the mass flow rate in forced circulation because of the underestimation of the void fraction in the riser. In particular, the average void fraction is half of the experimental one. To compensate for this limitation of the code, a double gas mass flow rate has been injected into the riser.

The coupled Ansys CFX and RELAP5 tool, adopted in WP4, has been realized through in-house developed FORTRAN routines that are called in specific "time-locations" of the Ansys CFX run (which is the master of the coupling process), and in turn they call specific executables written in Python to run and post-process RELAP5 results. The routines must guarantee the synchronization between the codes during the coupled simulation execution and the data exchange. The coupling strategy is based on a two-way connection, with a partitioned and sequential solution. The spatial domain approach is non-overlapping (or decomposition), while the time advancing scheme adopted in this analysis is semi-implicit.

For the coupled simulation, the RELAP5 model has been modified, removing the active part of the FPBS and inserting time-dependent volumes and time-dependent junctions, where the boundary conditions from the CFD calculation are imposed. The adopted approach is therefore the so-called "decomposition domain approach".

During the coupled run, the two-way exchange of information regards mass flow rate and temperature entering and exiting the two domains (CFX and RELAP5 ones) and the pressure drop computed by Ansys CFX. The instant at which the gas injection is stopped is assumed to be at t=0 s for both the RELAP5 and coupled simulations. The thermal power provided by the FPBS remains constant, as well as the water temperature and mass flow rate. The following comparison between the experimental, RELAP5 and coupled results during the transient involves only the integral parameters since local temperatures in the FPBS cannot be predicted by STH codes. Only the ADP10 results are reported. As shown on the left of Fig. 3, the LBE mass flow rate rapidly drops at ~1 kg/s as soon as the gas injection is stopped, and after ~1500 s (that is the simulation time considered for the coupled simulation) a new steady state establishes after the transition to the natural circulation regime.

The delay in the experimental mass flow rate drop-down can be attributed to the intrinsic delay in the measurement of the thermal flow meter and not to a physical delay, since the comparison between the temperatures has shown a good agreement. The curves of the coupled simulations almost overlap the RELAP5 stand-alone results since the transition is governed by phenomena that can be represented by a 1D model. In the analysed time frame, the coupled simulation does not show significant improvements in reproducing the experimental trends of the integral quantities. The small discrepancies can be addressed through a more detailed modelling of the FPBS and a different evaluation of the pressure drops in the component. The picture on the right of Fig. 3 shows a good agreement between experimental and numerical values of the FPBS inlet and outlet temperature, confirming the correct representation of the mass flow rate transition. Again, the numerical trends predicted by RELAP5 and the coupled tool are very close to each other, meaning that a 1D representation of the entire system in these conditions is sufficient for a proper description of the integral TH phenomena.

In this situation, the main advantage of applying the coupled tool is the possibility to compare the temperature variation in the FPBS TCs location in time with appropriate boundary conditions (coming from RELAP5), keeping at the same time a reasonable computational time.

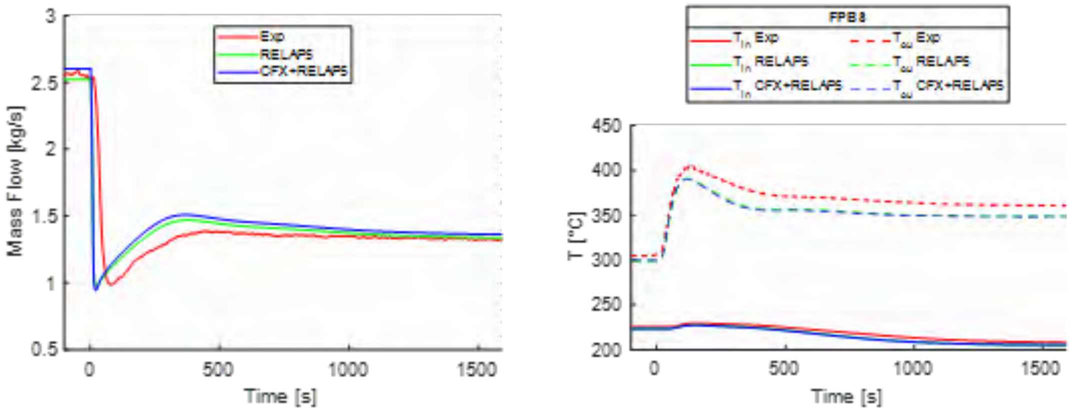

Fig.3: Experimental vs RELAP5 vs coupling mass flow rate (left) and FPBS temperature (right) comparison for ADP10 case.

Uncertainty quantification with sub-channel codes

Due to the high non-linearity and complexity of the problem, uncertainty quantification (UQ) in the thermal-hydraulic domain is typically done via Monte Carlo-based techniques. In the CRP, an innovative approach based on Generalized Perturbation Theory (GPT) has been attempted; the sub-channel level of detail seemed the right compromise for such an approach.

The variational form of GPT has thus been applied to the conservation equations, obtaining the so-called adjoint formulation, a necessary step to arrive at the definition of the sensitivity coefficients input to the UQ problem.

The adjoint equations have been preliminary implemented in the ANTEO+ code, in this first attempt, neglecting the coupling between the thermal and velocity fields. Sensitivity coefficients of bundle power and mass flow rate have been compared with values from exact perturbations, obtaining encouraging results.

Conclusions

The IAEA benchmark on NACIE-UP facility was proposed with the goal of evaluating numerical code capabilities in simulating HLM systems using experimental data collected by ENEA. The collected data allowed to characterize the NACIE-UP facility, with a great level of detail for the FPBS component. In the benchmark, ENEA was involved in all the work packages, obtaining good results with system, sub-channel, CFD and coupled CFD-system codes, proving the adequacy of models and methodologies that are usually used, but also of tools that have been specifically developed during the benchmark.

Acknowledgment

The work has been realized in the framework of the IAEA Coordinate Research Project-I31038, named "Benchmark of Transition from Forced to Natural Circulation Experiment with Heavy Liquid Metal Loop".

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FROM NEEDS TO DEVELOPMENT, VALIDATION AND APPLICATION (NEEDS): THE VIRTUOUS CYCLE FOR LFR CORE DESIGN

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Abstract

When dealing with the design of the core of a nuclear reactor, the availability of consolidated expertise, validated simulation codes, and consolidated design and analysis procedures, is a mandatory prerequisite for successful accomplishment.

Concerning lead-cooled fast reactors, the Laboratory for Design and Analysis of Nuclear Systems of ENEA holds a renowned leading position, acknowledged internationally. This derives from a longstanding experience, historically acquired on sodium-cooled fast reactors through the PEC, PRISM and BN-800 projects, and then shifted to systems cooled by heavy liquid metals within the context of Euratom research projects, national programmes and international collaborations.

Despite this, the Laboratory strives to pursue the continuous improvement of its assets, leveraging the activities on which it is engaged to focus the efforts on aspects of primary importance. The identification of the areas of improvement, and the definition of the priorities, are a direct result of the needs emerging as feedback from the activities performed. These needs, generally, involve an extension of the expertise in the pertinent domains (such as neutronics, thermal-hydraulics, thermo-mechanics, fuel performance) and its subsequent consolidation; the development of new codes (as was the case with ANTEO+, TIFONE and TEMIDE) and methods (such as the burnup sensitivity and uncertainty model for ERANOS), and their validation; the execution of new experiments (either by separate or integral effects) and their consequent assimilation.

Examples of the virtuous cycle of improvement of design capabilities, realized by the development of design building blocks, are presented with cases of practical application.

Keywords: *Core Design; Lead-cooled Fast Reactor; Research, Development and Qualification; Simulation and Modeling; Verification and Validation*

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Introduction

The Laboratory for Design and Analysis of Nuclear Systems of ENEA digs its roots in the Fast Reactor Department (Dipartimento Reattori Veloci, DRV), which, until 1987, was in charge – among others – of the design of the core of the PEC experimental reactor, which reached about 85% construction at the Brasimone Research Centre before the nuclear phase out in Italy. At that time, ENEA held the whole and sole responsibility for the design of the core of this reactor, relying on itself for all the competences and manpower for the purpose. Following the closure of this project, a relevant part of the expertise was lost. Despite this, a (small) group maintained determination to work on nuclear, transferring their skills to relevant projects within international collaboration endeavors. Relevant examples of this were the support to the core design of the PRISM reactor, pursued by General Electric in the USA (currently relaunched as NATRIUM with Bill Gates's TerraPower), and a major contribution to the study of a MOX-fueled option for the Russian BN-800 core.

Thanks to these initiatives, the competences were preserved until the late '90s, when – under the propelling action of Nobel laureate Carlo Rubbia, president of ENEA – the Energy Amplifier concept was launched, and followed by a number of collaborative projects on accelerator-driven subcritical systems (ADS), supported by a number of projects funded by the European Commission through its Framework Programmes. With the Energy Amplifier and the subsequent ADS projects, new competences were added, focused on the design of a fast reactor core cooled by a heavy liquid metal (HLM) such as lead or lead-bismuth eutectic (LBE). Despite the commonality of the fast spectrum, the primary cooling by HLM, indeed, changes the frame of technology and safety constraints which are to be accounted for in the design process, compared to those typical of sodium-cooled fast reactors (SFRs), according to the different physical and chemical properties of these liquid metals. It was, however, thanks to the sound expertise formed on SFRs, that the ENEA team managed to master the core design process for HLM-cooled reactors, as proven by the elaboration of ad hoc design processes (such as the so called "42-0" approach for ADSs [1], or the so called "adiabatic reactor" theory for critical systems [2]), and the leading role assumed both in Italian and European research projects, and within collaborations with HLM-cooled reactor vendors such as Westinghouse.

Core Design Methodology

The design of the reactor core – as that of any other system – moves from conceptual to basic to engineering (manufacturing) phases. At the same time – again, as for any system – it focuses on the left branch of the so called "V cycle" (see Figure 1), the one descending from mission to configuration, passing through the definition of objectives, goals and constraints and requirements while progressively detailing the macro into micro.

The right branch of the V cycle rises back from configuration to mission, passing through verification of requirements, constraints and objectives, extending the focus from micro to macro. Verification is aimed at highlighting any possible nonconformance to any of the requirements, objectives and constraints, or mission, the occurrence of which would result in the need for a corrective retrofit to the design, whose process would have to start again.

Therefore, it is the sake of design to be as accurate as possible, so to minimize the possibility of nonconformances, hence of rework. For this, the most appropriate and effective way – whose formalism is due to ENEA – is based on a holistic and predictive approach, which stands on reverting the whole verification problem so as to explicitly convert the objectives, constraints and requirements taken as input, into design parameters returned as output. Basing the configuration on values as close as possible to those obtained by this approach, would permit the design to score at first shot, minimizing (up to potentially zeroing) retrofits, and maximizing the profitability by minimizing the design margins towards the imposed limits.

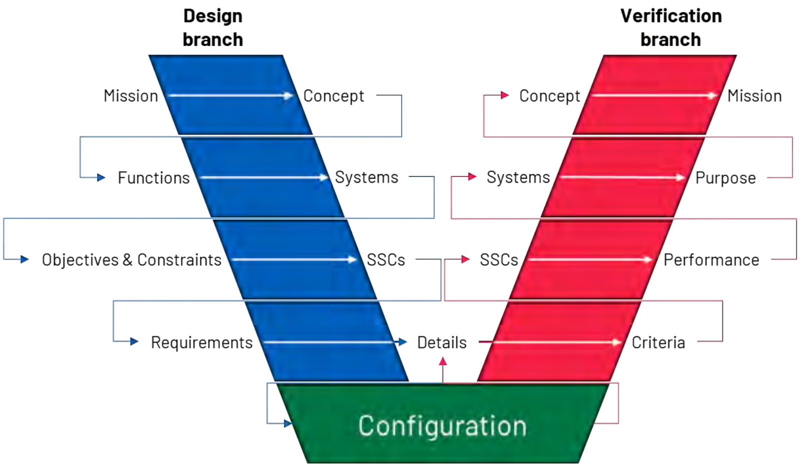

Fig.1: The "V cycle" for reactor design and verification.

Clearly, for this approach to be effective as aimed, two elements are essential: a sound database of experimental evidences, forming a reliable knowledge basis, and a complete portfolio of analytical codes, conceived to be design-oriented and validated upon the same database.

Experiments: the Knowledge Basis

Knowledge is mandatory. Without it, and as in this case, engineers would blindly attempt at designing systems, with the sole chance of stopping the trial-and-fail sequence, by finding a casual combination of things that works, but with no practical hope of making it optimal for the purpose. In science, knowledge stays in data, which comes from experimental results.

Data forms the basis for the quantitative statement of design criteria, which in turn materialize the requirements driving the design. Data also permits the development and calibration of models to reproduce the phenomena of interest, for use in calculation tools. Quality data, then, allows for minimising the uncertainty on phenomena, allowing for reducing margins for safety, hence broadening the design space, and possibly improving reactor performance.

Design-Oriented Codes: the Enabling Tools

The complexity of the problems to be dealt with in designing a reactor core requires the use of analytical tools. While the standard trend of development of such tools is to pursue better and better modelling capabilities, focusing on the micro-scale to capture the most elementary phenomena possible, design imposes rather different needs. The higher zooming capabilities generally imply longer running times, which are hardly compatible with the need for a quick feedback on the viability of an option, as is frequently requested by designers having to move from a blank sheet. Also, the pursuit for the microscopic, and multi-physics scales, generally adds complexity to the codes, hindering their ability to secure the traceability of input to output, i.e., sensitivity considerations, which is a precious information for a designer.

While these kinds of codes perfectly fit the needs for the verification phase of the "V cycle", others should be considered for the first phase, if specifically developed to be design-oriented.

Virtuous Cycle for Continuous Improvement

The PRO laboratory is deeply involved in design activities for liquid-metal-cooled fast reactors, and notably lead-cooled ones, through extensive collaborations at the national and international level. The exercising of its competences permits the continuous challenge of adequateness of the founding bases, and permits also to appreciate – with the variety of cases analyzed – the needs for extending the information already available, so as to enlarge as much as practical (and requested) the application domain of the competences.

A design activity concludes indeed with issuing technical specifications, which provide the reference values (and associated tolerances, in advanced design stages) for the manufacturing of all structures, systems and components of the core. All this information secures that the core systems behave in such a way to comply with the design criteria set as basis for the design itself. Criteria and specifications, thus, stand on the same knowledge data; data, which also form the basis for the validation of all the analytical tools used both in the design and verification phases. Completing and refining such a database is therefore the first and most urgent need for an effective design process, prioritizing the areas and topics to investigate on the basis of the outcomes of the design activities being performed.

Within the PRO laboratory, continuous support is given to the design and/or interpretation of experimental campaigns, providing new data on relevant subjects. This is the case of the newly conceived NETT1 Laboratory [3], the execution of new nuclear data measurements [4,5], or the assimilation of experimental data from NACIE experiments on wire- and grid-spaced rod bundles [6]. However, while some disciplines are relatively easy to investigate, allowing for new experiments to be arranged with limited use of resources – time, people and money –, other disciplines are more (in some cases, significantly) expensive to investigate. In such cases, the best use of past information is being made, recovering information from the available public databases available (such as the IRPhE, maintained by the OECD/NEA, for reactor physics experiments) or building a new one (such as the FRIDA database for fast reactor fuel performance irradiations) from sparse literature sources, provided the information contained therein is not fragmentary. All this data is used to generate the internal knowledge basis, and (in the short term) to validate the analytical codes in use by the laboratory.

The selection of cases and validation of the codes is based on sensitivity studies, which permit focusing the efforts on the phenomena and parameters of primary importance, so as to get the most benefit for a given effort. The same approach, based on sensitivity studies, also drives the development, or calibration, of design-oriented simulation codes. A complete software package is indeed pursued by the PRO laboratory, to permit the analysis of all phenomena occurring and ruling the behaviour of a liquid-metal-cooled fast reactor core. Following the same successful approach that made available ANTEO+ [7] (an updated version of a proprietary code for fuel assembly thermal-hydraulics), TEMIDE and TIFONE [8] (new codes respectively for fuel performance and whole core thermal-hydraulics), the current efforts involve, mainly, codes for the thermal-hydraulic analysis of a fuel assembly in deformed state (EFIALTE) or for the mechanical analysis of a fuel assembly and of the whole core (respectively, TEIA and FEBE); tools for data management and design optimization purposes (such as the DEMAGAGGER [9], meant at optimizing the core cooling groups); as well as methods extending the capabilities of existing codes to perform the whole spectrum of analyses requested to support and characterize a design (such as the combined deterministic-Monte Carlo method for the neutronic analysis of ADSs [10]).

On top of all this, the laboratory is pursuing the extension of its expertise and competences, aiming at covering all required domains as in its original scope (such as core mechanics and nuclear data evaluation), while consolidating and expanding the areas where an expertise is present and internationally recognized (such as reactor physics and core thermal-hydraulics).

Conclusions

The systematic critical review of the competences and tools that are available vs those that would be needed – as resulting from the extensive core design activities performed – is the key driver in the PRO laboratory for extending and improving its assets. The availability of these new capabilities, in turn, allows providing improved contributions to new projects, offering more chances for the critical review, hence self-growth, in a virtuous cycle towards consolidating a leading position in core design worldwide.

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ENABLING ADVANCED THERMAL-HYDRAULICS KNOWLEDGE: THE NETT1 LABORATORY

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Abstract

ENEA has gained a worldwide leading position in the field of Lead-cooled Fast Reactors core design and technology development. This was also possible thanks to the numerous facilities at the Brasimone Research Centre for the system-wide understanding of heavy-liquid-metals behaviour in complex geometries or challenging operating conditions.

As the design of LFRs advances, and with a particular focus on core design-related aspects, it is necessary to dive deeper into the local behaviour of components, to better understand the underlying physics and validate the relative software tools. This is particularly true for all components in which the hydraulic aspect plays a central role in their performances (e.g., lower plenum and fuel assembly interface).

Considering the peculiarities of heavy-liquid-metals and the ensuing experimental difficulties to measure very local hydraulic parameters, the PRO laboratory is devising a dedicated set of facilities by which the NETT1 laboratory (Laboratorio Nazionale Esperimenti di Termoidraulica e Tecnologia - 1) will be established. The facilities will be based on the hydraulic analogy and will work with water or water-like fluids to enable high resolution measurements of the velocity field and pressure profile inside complex-shape components via Particle Image Velocimetry and Matched-Index-of-Refraction techniques. While the starting focus will be LFR, thanks to the flexible nature of the laboratory, NETT1 will be able to support R&D&Q programs for other nuclear technologies, particularly advanced water-cooled Small Modular Reactors.

This paper presents the main features of NETT1 with details and ranges of possible experiments, as well as future steps to turn NETT1 into reality.

Keywords: *Experimental facilities; PIV; reactor core hydraulics*

Introduction

Lead-cooled fast reactors are one of the Generation IV technologies over which most of the industrial and research focus has been put by the research and industrial worlds alike. This has been possible

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thanks to the international efforts for understanding the behaviour of heavy liquid metals in terms of chemistry and thermal-hydraulics in complex geometries and challenging operating conditions.
ENEA, in particular, with the two Bologna branches, has achieved a leading position in this context thanks to a strong heritage in fast reactor design and a unique pool of experimental facilities located at the Brasimone Research Centre.
Trying to stay at the forefront of the expected international research activities of the future, the ENER-PRO laboratory has the ambition to maintain such leadership by envisioning a new laboratory for state-of-the-art experiments. Specifically, moving from the bulk expertise in core design, the next generation of needed experimental research has been pinpointed in the movement from system-wide to core-wide components behaviour in complex geometries. In such conditions, the hydrodynamics behaviour is instrumental for characterizing and assessing the overall component performance as well as validating the related software tools: a practical example is the lower core plenum and sub-assembly interface, which is characterized by a complex shift from low-to-high velocities.
This new experimental infrastructure, as an homage to the city of Bologna (where it will be located), has been called NETT1 [1] laboratory (Laboratorio Nazionale Esperimenti di Termoidraulica e Tecnologia - 1) with the main objectives of:

- hydraulic characterization of a wide array of components and test types targeting 1:1 scale when possible and
- validation of software tools and models, such as a turbulence model for Computational Fluid Dynamics software.

The NETT1 Laboratory

Moving from system-wide to local behaviour imposes a change of mindset, especially to what regards experimental techniques, possibilities and types of instrumentation. Heavy liquid metals are indeed strongly opaque and mandate the use of relatively high temperatures making the range of very local measurements possible only for the temperature field.
When of interest is the hydrodynamic field (velocities and pressures), the Reynolds-based hydraulic analogy can be leveraged by changing the reference fluid, thus allowing measurements not currently possible with heavy liquid metals, while preserving the underlying physical behaviour of the component under testing. Typical fluids for such purpose are water or other water-based fluids with specific optical properties (see Section "Measurement techniques").

[1] The one ("1") in the name highlights the ambition of this laboratory to be the first of its kind and the auspices to be followed by many other upgrades.

Capabilities

The main target capabilities can be summarized as related to an isothermal laboratory to collect hydraulic measurements, like:

- integral measurements of pressure losses and mass flow, and
- local measurements of velocity fields.

To this extent, two enveloping types of tests have been taken as reference for designing the facility:

1. a fuel assembly of the ALFRED reactor [1], considering the pin bundle section where the Reynolds' number is the highest (around 40000) and
2. the inlet plenum region of ALFRED [1] connecting the downcomer of the pool with the sub-assembly inlet foot, with a Reynolds number locally above 100000.

The two experiments cover high velocity situations in tight spaces and low velocities in wide regions, thus providing different sets of experimental challenges. In Table 1, some of the main parameters of the two experiments are collected. As can be seen, the two reference set-ups are remarkably different in terms of macroscopic scale (subchannel vs. wide region) and velocities; the volumetric flow rate is also an order of magnitude higher for the inlet plenum, so high in fact that a 1:8 re-scaling (always preserving the hydraulic analogy) is foreseen to be necessary to not unreasonably increase the pumping requirements (e.g., power) (see Section "Envisaged layout").

Table 1: Main parameters of the reference test sections.

Parameter	**Fuel Assembly**	**Inlet plenum**
Reynolds number [-]	40000	>60000
Characteristic scale [m]	≈0.1	≈1
Velocity (water) [m/s]	4.5	1.5
Turbulence scale [μm]	1	N/A
Turbulence time [μs]	10	N/A
Volumetric flow rate [m^3/h]	150	3000

Measurement techniques

The most challenging experimental component of the laboratory is the Time Resolved Particle Image Velocimetry (TR-PIV) [2] apparatus, which enables the high-resolution measurements of the velocity field. As depicted in Figure 1, the principal components are:

- particles in suspension in the flowing fluid (e.g., alumina particles);
- a laser and related optics to illuminate a given region of space of interest, whose light impinges on the particles;
- a high-speed (greater than 1 kHz), high-resolution camera (tens of microns) for capturing local and instantaneous movement of the suspended particles via the reflected laser light;
- a laser-camera synchronizer to ensure camera is "light-taking-mode" when laser is in "pulsing-on-mode";
- a post-processing software that collects the images of the camera and deduces the velocity field via differential movement of the captured particles.

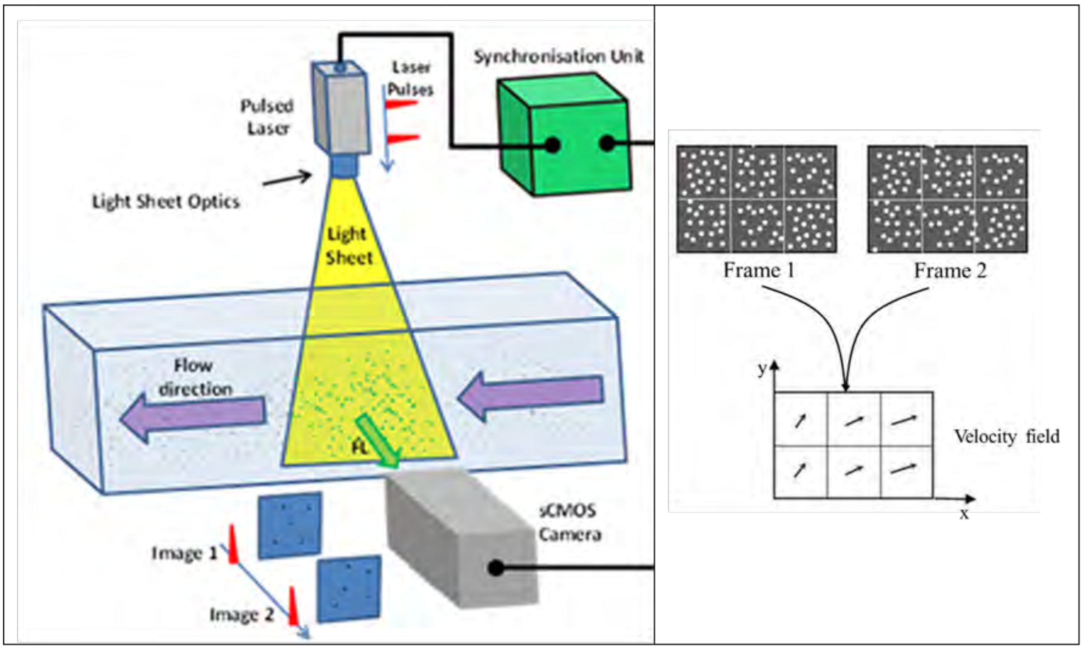

Fig.1: Representation of a typical PIV setup (on the left) and post-processing operations to retrieve the velocity field (on the right) (adapted from [3]).

To this end, the flowing fluid must be transparent to the laser light to easily transmit it to the particles and back to the camera. Also, it is important, especially in cases in which information on the velocity field must be collected deeper in the test section, that the fluid has an index of reflection equal to the solid structure internal to the test section so to not distort the path travel by light and thus to not introduce noise in the measurement process. This latter feature is often called the Matched Index of Refraction (MIR) technique [2]. An example of a typical result that can be obtained from a PIV-MIR measurement is shown on the right of Figure 1.

Envisaged layout

The Piping and Instrumentation diagram of the laboratory is shown in Figure 2 where the main components are visible:

- the test section part (see Section "Capabilities");
- the circulating pump (PC-101);
- the expansion vessel (S-101);
- the discharge tank (S-102) and related lines;
- the feedwater line;
- the temperature control system.

The feed and discharge lines are necessary to introduce the fluid with the particles in the system and then remove it for storage and recycling, for cost-saving considerations. The selected circulating pump has a nominal power of 18 kW and can reach around 360 m^3/h of volumetric flow for water and water-like fluids (based on Table 1). The expansion vessel is instead necessary to allow for absorbing temperature fluctuations in the loop without exceeding the pump and piping limits of 10 bar.

A bypass line to the test section with a regulating valve is also present to allow a control on the mass flow on the test section without changing the pump working point (i.e., without penalizing efficiency). Instrumentations for control of the loop, like flow meters, pressure transducers and thermocouples are also depicted in Figure 2.

The necessity of a temperature control system in an isothermal facility has been evaluated based on the heat rejected by the pump (efficiencies between 60% and 80%, depending on the operating point) and under conservative assumptions of all heat taken by the fluid and adiabatic loop. The analysis has revealed that the dependence of the water viscosity on temperature in experiments lasting several minutes can already induce significant deviations on the Reynolds number thus introducing an important source of errors in the boundary conditions of the experiment. Furthermore, when MIR is employed the dependency on temperature of the refraction index can be another source of error further contributing to the need for a tight temperature knowledge.

For the above reasons a temperature control system has been added and dimensioned based on a finned air cooler to contain the required tubing length (less than 1.5 m) and thus system volume, while allowing a wide range of water temperatures in the loop (between 30 °C and 90 °C).

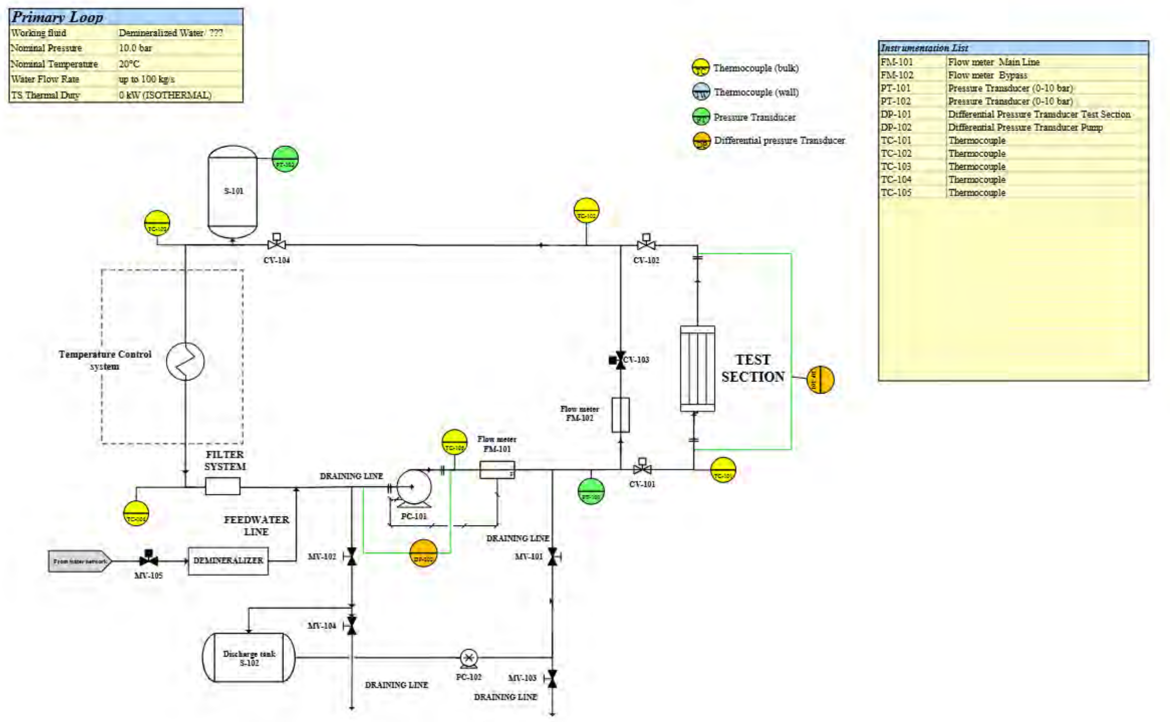

Fig.2: Piping and Instrumentation diagram of the NETT1 laboratory.

Conclusions

The NETT1 laboratory is a stepping stone for allowing the ENER division to project its leadership in core design of LFR systems into the future, expanding experimental capabilities and software validation domains.

The groundwork for arriving at a preliminary layout of the laboratory, which fulfils the declared objectives, has been completed, including components required, volumes to be allocated and estimated costs. The next step is the definition of the hosting building and the selection of the laser-camera combination that fits the identified necessities.

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ENHANCING LFR ANALYSIS ACCURACY: REFINING NUCLEAR DATA BY DIFFERENTIAL AND INTEGRAL MEASUREMENTS

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Abstract

The development of advanced reactors, particularly Lead-cooled Fast Reactors (LFRs), depends on high-quality nuclear data to enable precise and accurate simulations, as well as the safe and efficient design of systems - especially given the limited operational experience available. ENEA's Laboratory for Design and Analysis of Nuclear Systems is actively addressing this need through a newly launched program aimed at enhancing its analytical capabilities, leveraging the TAPIRO research reactor. TAPIRO has always been a key asset in fast reactor research, supporting a wide range of studies, including neutronic studies, code validation, component qualification, and detector calibration. To reassert its strategic role, feasibility studies have recently been conducted for new experiments focused on neutron propagation phenomena relevant to LFRs. Sensitivity and uncertainty analyses using ERANOS and MCNP codes revealed the significant influence of copper cross-sections, particularly those of ^{63}Cu and ^{65}Cu, on the quality and representativeness of results, highlighting the pressing need for improved nuclear data. In response, the RAMEN (RAME + n) initiative was proposed within the Euratom-funded APRENDE project, in collaboration with the n_TOF Collaboration at CERN. This effort is underscored by the inclusion of proposals for new measurements of three copper reaction channels (capture, elastic, inelastic) in the NEA's High Priority Request List (HPRL).

Keywords: *Cross-sections; Fast Reactors; Nuclear data; Sensitivity Analysis; TAPIRO*

The strategic role of TAPIRO for advanced reactor development

The development of advanced nuclear technologies—notably Lead-cooled Fast Reactors (LFRs)—offers significant advantages in terms of inherent safety, improved fuel utilization, and more effective management of long-lived radioactive waste. This requires a deep understanding of neutron transport and reactor physics, which strongly depend on high-quality nuclear data.

While substantial progress has been achieved in the engineering domains of fast reactor development, such as thermal-hydraulics and materials science, a significant gap persists in the experimental

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database related to LFR neutronics. This is particularly evident in the qualification of prototypical reactor components and materials under fast-spectrum conditions.

Longstanding tradition in fast reactor research, focused on LFR technologies, is a key element of the TAPIRO research reactor, located at ENEA Casaccia Research Centre near Rome.

TAPIRO is a compact, zero-power research reactor, and its cylindrical core is made of metallic uranium and molybdenum alloy, highly enriched in ^{235}U isotope. It is cooled by flowing helium, and a neutron fast spectrum is attained inside the core, being absent any form of moderation. The reflector is made of copper, and the control devices are copper rods, being part of the reflector itself. Their insertion or extraction causes the neutron reflection to be increased or decreased, respectively.

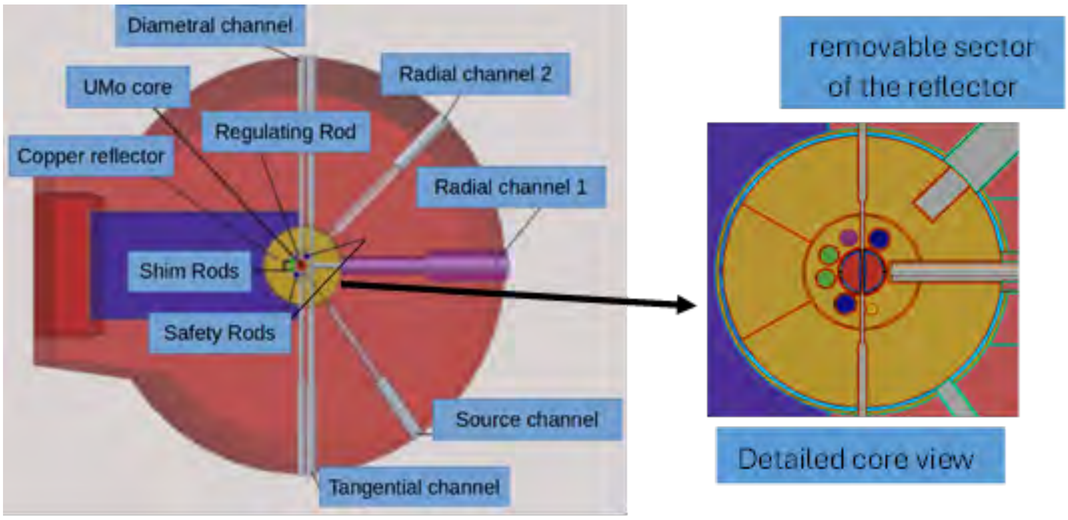

Fig.1: The TAPIRO reactor cross-section: core, irradiation channels and control rods.

The TAPIRO facility can produce a broad range of neutron spectra across multiple irradiation channels. Near the core, it delivers a quite pure fast fission spectrum (1 keV–10 MeV), peaking around 820 keV with an average energy between 1–2 MeV, ideal for high-precision integral experiments, supporting nuclear data validation and code benchmarking.

Designed to provide an intense, flexible fast neutron source, TAPIRO has long focused on integral experiments, particularly for refining cross-section data, qualifying detectors and components, and validating computational tools for fast reactor applications.

TAPIRO experiments supporting the design of innovative systems

Experimental activities are under development to support the design of innovative nuclear systems, such as LFRs and accelerator-driven systems (ADS). The objectives are to inspect minor actinides nuclear data through both integral and differential irradiations and neutron flux propagation in lead to study neutron interaction with this coolant for innovative reactors.

Concerning the study of advanced fuels and radioactive waste incineration, TAPIRO reactor has contributed to the international AOSTA campaign, which demonstrated the feasibility of irradiating minor actinides to improve nuclear data by means of OSMOSE and IRRM irradiation samples. OSMOSE samples consist of a natural uranium dioxide (UO_2) matrix pellet encased in double zircaloy sheaths. They contain dispersed minor actinides (including ^{237}Np, ^{242}Pu, ^{241}Am, and ^{243}Am). OSMOSE samples are intended for insertion into the Tangential and Radial-1 channels. Instead, IRRM samples feature a ^{241}Am oxide pellet in the chemical form of Al_2O_3, contained in aluminium cladding. They can be placed in the central part of the Diametral channel, in the Tangential and Radial-1 channels.

Experimental campaigns oriented to LFR technology development have been performed, considering the ALFRED reactor, the candidate LFR demonstrator within the European framework. Preliminary studies regarded neutron propagation across a portion of the TAPIRO reflector – a 60° removable sector – that was substituted with a block of nuclear-grade lead.

Both Monte Carlo (MCNP) and deterministic (ERANOS) models were developed to calculate and to compare TAPIRO and ALFRED neutron spectra at some positions of interest and for a set of observables.

In addition, lead blocks were introduced downstream of the removable sector to improve the representativeness of TAPIRO.

Differences between the ALFRED and the TAPIRO spectra were observed upstream of the lead block in the fast energy range (1 keV to 1 MeV). By contrast, downstream of the lead block the spectra of both reactors were found to be superimposable.

Preparation studies for both experimental campaigns involved simulations to evaluate the impact on effective multiplication factor (k_{eff}) of the system and therefore the spectral indices ^{238}U/^{235}U and ^{237}Np/^{235}U (hereafter f28/f25 and f37/f25). They characterize the neutron energy spectrum and represent the relative rates of reactions with specific isotopes under the neutron flux, providing insights into the neutron spectrum for each application.

Evaluation have been obtained with ERANOS code, using the latest ENDF/B-VIII.0 nuclear data prepared in ECCOLIB (cell code) format. A dedicated covariance matrix, extended with copper data from JEFF-3.3 - the only recent library providing complete copper covariances - was used [1]. Sensitivity and uncertainty (S&U) analysis have been carried out as well through the ERANOS suite. f28/f25 and f27/f25 spectral indices were used to evaluate the spectrum softening due to the propagation through the lead block.

Impact of copper cross-sections on TAPIRO simulations and analyses

ERANOS simulations revealed that ^{63}Cu and ^{65}Cu neutron capture, elastic and inelastic channels are amongst the major contributors to sensitivity and uncertainty, both for k_{eff} as well as for the f28/f25 and f37/f25 spectral indices.

For k_{eff}, the primary contributor is ^{235}U fission, but ^{63}Cu and ^{65}Cu elastic and inelastic reactions contribute up to 12% and 6%, respectively.

For f37/f25, ^{237}Np fission is dominant, followed by ^{63}Cu elastic and inelastic reactions (67% and 41%), with ^{65}Cu elastic contributing 33%.

For f28/f25, the ^{63}Cu inelastic reaction is the top contributor (100%), with additional influence from its elastic (54%) and capture (39%) reactions. ^{65}Cu reactions are less significant, since less abundant than ^{63}Cu. In terms of uncertainty, ^{235}U fission remains the main source for k_{eff}, but ^{63}Cu inelastic and elastic reactions contribute notably (26% and 25%), followed by ^{65}Cu elastic (19%) and inelastic (10%).

For f37/f25 and f28/f25, ^{63}Cu inelastic reactions are the major uncertainty drivers (34% and 76%, respectively), with ^{65}Cu inelastic contributing 13% to f28/f25. Elastic and capture reactions contribute less than 16% and 13%.

Representativeness analysis for TAPIRO concerning the ALFRED core [2] was performed for the spectral indices f27/f25 with and without the contribution due to copper isotopes to the covariance matrix (results shown in Table 1) [3].

Table 1: Representativity coefficients without data of the Copper isotopes.

Representativity f27/f25	**With Copper**	**Without Copper**
TAPIRO/ALFRED downstream	0.53	0.64

Results show that uncertainties on copper cross-sections can hinder the potential representativeness of TAPIRO. High-quality copper cross-sections could enhance the significance of such experiments.

Additional evaluations have been performed by means of the MCNP Monte Carlo code to further assess the impact of copper nuclear data on TAPIRO criticality, for both ^{63}Cu and ^{65}Cu isotopes. For this purpose, the multiplication coefficient (k_{eff}) has been evaluated with direct calculations and through sensitivity analyses.

A reference critical configuration has been considered with the entire ENDF/B-VIII.0 data library. Thus, the single ^{63}Cu or ^{65}Cu isotope has been substituted, taking the equivalent data file from different libraries. All the other isotopes were, at each run, maintained from the ENDF/B-VIII.0 data set.

As expected, the system revealed its sensitivity to the interactions between neutrons and copper. All results are presented in Table 2. Criticality calculation has also been done with the ENDF/B-VIII.1 data library.

Table 2: Impact of copper nuclear data on TAPIRO multiplication coefficient (keff).

Reference ENDF/B-VIII.0 k_{eff}	**1.00002**	**[all results with at most +/- 8 pcm as 1 std]**	
Ref. + ^{63}Cu (JEFF3.3)	+637 pcm	Ref. + ^{65}Cu (JEFF3.3)	-20 pcm
Ref. + ^{63}Cu (JEFF4T3)	+316 pcm	Ref. + ^{65}Cu (JEFF4T3)	-229 pcm
Ref. + ^{63}Cu (JENDL-5)	+147 pcm	Ref. + ^{65}Cu (JENDL-5)	-218 pcm
Ref. + ^{63}Cu (TENDL-21)	+102 pcm	Ref. + ^{65}Cu (TENDL-21)	+17 pcm
Ref. + ^{63}Cu (ENDF/B-VIII.1b)	+218 pcm	Ref. + ^{65}Cu (ENDF/B-VIII.1b)	-47 pcm
Ref. + ^{63}Cu (ENDF/B-VIII.1)	+340 pcm	Ref. + ^{65}Cu (ENDF/B-VIII.1)	-196 pcm
Complete ENDF/B-VIII.1	+129 pcm		

In addition, sensitivity analyses have been performed by MCNP as well, with both ENDF/B-VIII.0 and ENDF/B-VIII.1 nuclear data sets. Splitting contributions to k_{eff}, possible by linear perturbation, copper isotopes confirmed their relevant role concerning elastic scattering, inelastic scattering and absorption.

New measurements of copper cross-sections at n_TOF facility

To further highlight the availability of actual copper cross-section data, a comparison among the most recently evaluated libraries – ENDF/B-VIII.0, JEFF-3.3, and JENDL-5.0 – was conducted [4]. Significant discrepancies were found, particularly for (n,γ), (n,n), and (n,n′) reaction channels. For the (n,n) reaction, uncertainties are highest for ^{65}Cu (up to 15%), with discrepancies of 10% among libraries in the MeV range. For the (n,n′) reaction, experimental data are extremely scarce, with uncertainties ranging from 10% to 25% up to 2 MeV. Evaluations diverge by up to 25% for ^{65}Cu.

ENEA submitted an official proposal to carry out new measurements for the ^{63}Cu and ^{65}Cu isotopes, within the n_TOF International Collaboration and approved by CERN ISOLDE and Neutron Time-of-Flight Committee (INTC) [5]. The approved measurement campaign includes (n,γ) measurements, performed in summer 2024, as well as upcoming transmission measurements, scheduled in June 2025. Elastic and inelastic scattering cross-sections setups are being developed with the aim of an additional (n,n) and (n,n’) measurement to be possibly performed in 2026.

Within the APRENDE framework (Addressing Priorities of Evaluated Nuclear Data in Europe), ENEA proposed the RAMEN initiative (from RAME, copper in Italian, and *n* for neutrons), carried out and developed in collaboration with INFN (Italian National Institute for Nuclear Physics). Thanks to the n_TOF Collaboration, the state-of-the-art neutron time-of-flight facility at CERN is available to perform measurements across a broad energy spectrum (eV to several MeV). The RAMEN initiative fostered the submission of three targeted requests to the High Priority Request List (HPRL) of the OECD Nuclear Energy Agency (NEA), a key mechanism through which the international nuclear data community identifies and ranks the most critical data needs. Each submission addresses a specific copper reaction channel: capture, elastic, and inelastic scattering.

Conclusions

Within the pathway to enhance the innovative nuclear reactors, TAPIRO research reactor is a key irradiation facility to improve the understanding of both irradiation behaviour of actinides concerning radioactive waste management issues and neutron interaction with lead to support LFR design.

Feasibility studies and preparatory simulations have been conducted to demonstrate the feasibility of the AOSTA irradiation campaign to inspect minor actinides burnup properties with both OSMOSE and IRRM test samples.

On the other hand, neutron-lead interaction is supposed to be studied by means of a lead block insertion replacing a portion of the TAPIRO reflector, taking advantage of a neutron spectrum representative of fast reactors.

Setup preparation highlighted the critical role of precise and accurate copper cross-sections since the TAPIRO reactor is particularly sensitive to this element, constituting its large reflector.

To support the development of nuclear systems such as LFRs using TAPIRO, the RAMEN initiative - part of the Euratom APRENDE framework and in collaboration with CERN's n_TOF Collaboration - has been launched by ENEA, and a new campaign focused on high-precision measurements of copper cross-sections is ongoing, both for ^{63}Cu and ^{65}Cu isotopes.

This project, which has already gained recognition through its inclusion in the OECD/NEA High Priority Request List (HPRL), seeks not only to generate crucial new experimental data but also to strengthen Europe's expertise in nuclear data evaluation.

Acknowledgment

Authors acknowledge the Italian National Institute for Nuclear Physics (INFN), Prof. Cristian Massimi (Dept. of Physics, University of Bologna) and the n_TOF Collaboration at CERN.

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SESSION 3

ADVANCED TECHNOLOGIES FOR FUTURE NUCLEAR APPLICATIONS

Chairperson: *B. Ferrucci*

Session 3 was aimed at providing an overview of a diverse yet coherent set of activities addressing innovative technological developments for present and future nuclear applications. The session highlighted how nuclear science and engineering competences are increasingly extending beyond traditional power generation, contributing to advanced medical applications, space exploration, particle accelerator technologies, and the digital transformation of nuclear system analysis.

Several contributions illustrated the application of advanced modelling and simulation tools to non-conventional nuclear fields. In particular, the use of Monte Carlo techniques for radiation transport and dosimetry was presented as a key enabler for the development of innovative medical applications, demonstrating how methodologies originally developed for reactor analysis can be effectively transferred to the optimization and assessment of radiation-based therapies.

Other presentations addressed the role of nuclear technologies in space applications, focusing on the design of compact, reliable, and high-performance systems capable of operating in extreme environments. In this context, nuclear energy was presented as a strategic solution for long-duration missions and extraterrestrial infrastructures, where high energy density, reliability, and autonomy are essential requirements. The discussed activities highlighted the multidisciplinary challenges associated with these applications, spanning reactor design, system integration, safety, and energy management.

The session also included contributions dedicated to advanced technologies for particle accelerators, with particular emphasis on the design and analysis of beam-intercepting devices based on heavy liquid metals. These activities showcased the strong links between nuclear engineering, fluid dynamics, materials science, and high-energy physics, addressing complex phenomena such as intense energy deposition, shock wave propagation, and thermo-mechanical effects under extreme operating conditions.

Finally, the growing role of digitalization and data-driven approaches in nuclear engineering was highlighted through the application of machine learning techniques to improve the robustness and efficiency of numerical simulations. The integration of artificial intelligence methods with established severe accident and thermal-hydraulic codes was presented as a promising direction to enhance convergence, reduce computational costs, and support advanced analysis workflows.

Overall, the session demonstrated how the continuous evolution of nuclear technologies relies on the integration of advanced modelling tools, innovative experimental and computational approaches, and cross-disciplinary expertise. The presented activities clearly reflected the capability of ENEA and its partners to contribute to a broad spectrum of future nuclear applications, reinforcing the strategic role of nuclear technologies in energy, health, space, and advanced scientific infrastructures.

MCNP SIMULATION OF THE INTRA-OPERATIVE RADIATION THERAPY (IORT) TREATMENT OF SOLID CANCERS WITH FAST NEUTRONS AND COMPARISON WITH THE DOSIMETRY PARAMETERS OF STANDARD IORT TECHNIQUES ADOPTING X-RAYS AND ELECTRONS

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Abstract

Two prototypes of a compact neutron generator (CNG), provided by the TheranostiCentre company, are installed in a new equipped laboratory at the ENEA research centre in Brasimone (BO) built with funding of the Emilia Romagna Region. An ambitious application of the CNG - producing fast neutrons through the DD fusion reaction - is represented by the neutron Intra-Operative Radiation Therapy (nIORT®) aiming the adjuvant treatment of solid tumours. In this perspective, the Monte Carlo code MCNP6.1 was used to model the CNG coupled with a typical cylindrical IORT applicator to be inserted into the surgical cavity. The brain tumour irradiation (after craniotomy) was simulated with a 4-cm-diameter applicator, used also to simulate the standard IORT treatments with X-rays (50 keV energy) and electrons (1 MeV).

Thanks to the high flux (~10^8 cm^{-2} s^{-1}) of fast neutrons having a high Relative Biological Effectiveness (RBE) in cancer cells' killing, the CNG is potentially able to administer the common clinical dose targets of 10÷20 Gy(RBE) within about 8÷16 minutes. The comparison between the dosimetry parameters demonstrated that the neutron beam offers complementary features respect to standard treatments with focused beams of X-rays and electrons for both:

1- the diffused dose profiles in superficial tissues, allowing to irradiate extended tumour beds with topographic irregularities and normally filled by quiescent cancer cells;

2- the dose profile in tissues depth, that is not steep as the electrons' dose profile and less penetrating than the X-rays' one, with potential benefits for the tumour local control.

Keywords: *Compact Neutron Generator (CNG); Fast neutrons; Intra-Operative Radiation Therapy (IORT); MCNP Monte Carlo code; Dosimetry parameters*

Introduction

This research study was performed in collaboration with the TheranostiCentre Srl company (TC) and carried out in parallel with the LINCER project [1]. The most significant outcomes of the last 5-year activities are represented by the installation of two prototypes of an innovative Compact Neutron

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Generator (CNG, provided by TC) in a new bunker-laboratory at the ENEA Research Centre in Brasimone (BO). A challenging application of the CNG - creating neutrons of 2.45 MeV energy through the Deuterium-Deuterium (DD) fusion reaction - is represented by the neutron Intra-Operative Radiation Therapy (nIORT®) technique [3].

The nIORT® basic idea is to irradiate the surgical cavity with fast neutrons, having a high Linear Energy Transfer (LET) and thus offering superior biological effectiveness in cancer cells' killing when compared to standard IORT techniques exploiting low-LET particles. Indeed, high-LET particles generate fewer but larger double-strand DNA breaks per unit dose when compared with low-LET particles as X-rays, typically inducing single-strand DNA breaks which can be efficiently repaired by cancer cells. The Relative Biological Effectiveness (RBE) of fast neutrons (at ~2.5 MeV energy) was measured in the @3.5÷5 range for tumour cells and @2.5 for normal tissues [4], when compared with the photon and electron radiation used in standard IORT treatments having a unitary RBE value.

The CNG performances for potential nIORT® treatments were evaluated for the irradiation (with craniotomy) of brain cancers, among which the glioblastoma multiforme (GBM) represents one of the most aggressive and mortal pathologies. The simulations - carried out with the Monte Carlo N-Particle (MCNP) ver. 6.1 code [5] in coupling with ENDF/B-VIII.0 nuclear data [6] - were devoted to calculating the main dosimetry parameters in the superficial tissues of the surgical cavity and in brain depth. For this purpose, the CNG was modelled as equipped with the typical IORT applicator that, in standard IORT modalities, is a pipe tube to be inserted in the surgical cavity: a cylindrical applicator of 4 cm in diameter was adopted in this study (see Section "MCNP model of the CNG with IORT applicator").

The dose profiles obtained by the potential nIORT® irradiation of the brain surgical cavity were evaluated (see Section "Simulations with fast neutrons"). The same dosimetry parameters were calculated with MCNP simulating the standard IORT irradiation treatments adopting focused beams of X-rays and electrons (see Section "Simulations with 50 keV X-rays and 1 MeV electrons"). The comparison among the dose profiles highlighted the complementary features of the nIORT® technique respect to the standard IORT treatments (see Section "Clinical-equivalent dose profiles").

MCNP model of the CNG with IORT applicator

The CNG design essentially consists in a hermetic void tube in which D+ ions are accelerated against a titanium target, in which the DD fusion reaction occurs. The acceleration column made of High-Density Polyethylene (HDPE, having excellent properties in shielding neutrons) is shown in the 2D section of the MCNP model reported in the left part of Fig. 1, surrounded by an external shield made of borated PE and a layer of lead (mainly for g rays).

The right side of Fig. 1 shows an enlarged drawing of the MCNP model of the cylindrical IORT applicator (4 cm in diameter, purple colour), shaped around the HDPE bearing-ledge structure (or "nose", yellow) containing the titanium target (orange) on which the 2-cm-diameter ion beam impinges.

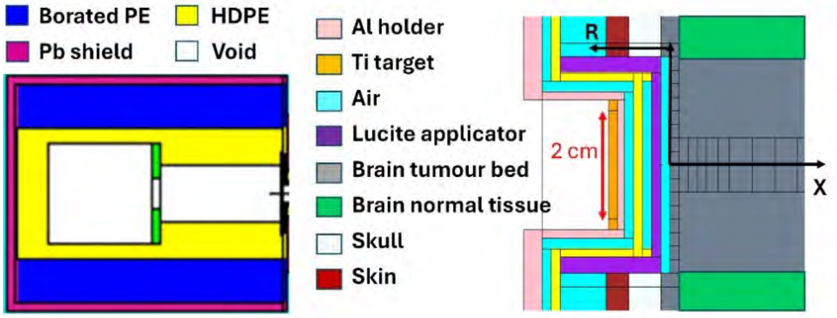

Fig. 1: MCNP model of the CNG and surrounding shields (left). Zoom on the MCNP model of the 4-cm-diameter cylindrical IORT applicator inserted in the brain surgical cavity (right).

For simulating the brain irradiation with craniotomy, the IORT applicator (about 2 cm long) was positioned in the skull "opening" surrounded by skin and skull tissues. In real cases, the tumour bed extension is not so well-delimited as in the MCNP model of Fig. 1: however, the flux and dose rate levels were accurately calculated / monitored in small volumes for all the tissues around the IORT applicator and in brain depth. To be noticed the "X" and "R" axes shown in the right part of Fig. 1, in which (see also Fig. 2 with dose profiles):

1. the "X" axis refers to the calculation points in deep brain (surface at X=0);
2. the "R" axis represents the distance between the centre of the tumour bed (R=0) and centre of each cell representing surface tissues of the surgical cavity around the IORT applicator. With the 4-cm-diameter cylindrical shape, R corresponds to the brain cells in front of the (circular) applicator endcap until 2 cm, while bigger R values refer to the skull and skin cells on its lateral side.

Simulation of IORT treatments

Simulations with fast neutrons

The DD-CNG generating 2.45 MeV neutrons from the titanium target with an almost isotropic direction of emission was simulated with MCNP. The calculations were not restricted to neutrons (*i.e.*, the primary ones from the titanium target and the secondary ones coming from the scattering with the CNG walls and into the biological tissues), but they also include the photons: *i.e.*, gammas (g) created by neutrons interaction with matter. The physical dose rates into biological tissues due to neutrons ($D'_{f,n}$) and gammas ($D'_{f,\gamma}$) were calculated by MCNP.

Traditionally, dose in proton radiotherapy is prescribed as Gy(RBE) by scaling up the physical dose (Gy) by 10% (*i.e.*, RBE = 1.1). Similarly, the nIORT® "clinical-equivalent" dose rate due to neutrons and gammas ($D'_{eq,tot}$) was calculated from the physical dose rate results by:

$$D'_{eq,tot} = RBE_{\gamma} \cdot D'_{f,\gamma} + RBE_{n} \cdot D'_{f,n} \qquad (1)$$

where:

1. the RBE of fast neutrons (RBE_n) was assumed 3.5 in all biological tissues (as for cancer cells, while it is limited to 2.5 in normal tissues [4]);
2. the RBE value of secondary photons in biological tissues (RBE_{γ}) was assumed unitary.

Starting from the clinical-equivalent dose target – usually named clinical endpoint ($D_{eq,T}$), typically 10÷20 Gy(RBE) - the Treatment Time (TT) can be easily deduced by the ratio:

$$TT = D_{eq,T} \;/\; D'_{eq,tot} \qquad (2)$$

The spatial distributions of the clinical-equivalent dose rates (1) in the surgical cavity and in tissues depth were accurately calculated by MCNP to estimate the peak value administered at the centre of the tumour bed's surface - usually corresponding to the $D_{eq,T}$ clinical end point - as well as the dose administered in the surrounding tumor bed margins and in normal tissues.

The CNG operating with a 3.3 10^9 s^{-1} yield can administer a physical dose rate peak of 0.35 Gy/min (at the centre of the tumour bed, mostly due to neutrons) and a clinical-equivalent dose rate peak of about 1.22 Gy(RBE)/min, as derived by equation (1). Therefore, the common clinical dose targets of 10÷20 Gy (RBE) could be administered in @8÷16 minutes.

Simulations with 50 keV X-rays and 1 MeV electrons

In the MCNP simulations of the standard IORT treatments with X-rays of 50 keV energy [7] and 1-MeV electrons [8], the source particles were generated directly on the external (circular) surface of the applicator endcap, with a focused beam of 2 cm in diameter. Referring to the right part of Fig. 2, X-rays and electrons were generated in the central region of the IORT applicator endcap and crossed only a 0.2 cm air gap before reaching the brain tumour bed.

About the physical interaction of these irradiation particles in the biological tissues and their MCNP simulation, it can be noticed that:

1. X-rays have a significant power of penetration since the secondary electrons produced deposit their energy over large distances in the tissues. Coherently, the MCNP calculations were not restricted to the primary photons but considered the secondary electrons and their contribute to the physical (D'_f) dose rates.

2. Travelling through a medium, electrons interact with atoms by a variety of processes owing to Coulomb force interactions. In low-atomic-number media such as water or tissues, electrons lose energy predominantly through ionization and excitation events (*i.e.*, inelastic collisions with atomic electrons).

For both 50 keV X-rays and 1 MeV electrons, the RBE value was assumed one: thus, the clinical-equivalent dose (D'_{eq}) numerically coincides with the physical one (D'_{f}).

Clinical-equivalent dose profiles

The dosimetry parameters obtained by MCNP refer to brain cancer treatment, but the results could be almost generalized to other severe solid tumors in different organs and areas of the body. Unlike standard RT techniques relying on focused beams of X-rays or electrons, the diffuse neutron beam can treat large and topographically irregular tumor beds. This peculiarity appears evident in the left part of Fig. 2 showing the dose rate profiles in the superficial tissues of the surgical cavity, normalized to the same clinical-equivalent dose peak at the center of the tumor bed. The treatment of extended and irregular tumour beds represents a therapeutic challenge with existing IORT technologies with a focused beam of X-rays or electrons, whereas the diffuse neutron beam could allow to irradiate the tumour bed margins, normally filled by quiescent cancer cells, with potential benefits for the tumour local control.

The right part of Fig. 2 shows the dose profiles in brain depth. The nIORT® dose gradient result to be not steep as the electrons' profile and less penetrating than the X-rays' profile. Thus, the nIORT® device could supply "complementary" features respect to the standard IORT (and more generally RT) treatments for its peculiar dose profiles in both superficial and deep tissues.

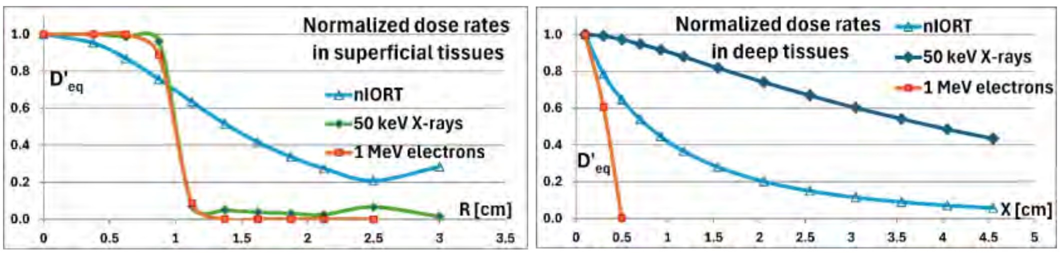

Fig. 2: MCNP results for normalised clinical-equivalent dose rates in the superficial tissues of the brain surgical cavity (left) and in brain depth (right) with different IORT techniques.

The diffuse neutron beam should allow to irradiate extended tumour beds with margins potentially filled by quiescent cancer cells, as the GBM representing the most common and deadliest primary malignant brain tumor in adults for its rapid growth, high invasiveness and remarkable resistance to conventional therapies. The large tumor bed (~4÷10 cm) and its diffuse infiltration into surrounding tissues makes the complete surgical resection nearly impossible and, even with maximal surgical removal, residual microscopic disease typically persists contributing to high recurrence rates, frequently observed within 2÷3 cm from the initial GBM lesion. These local recurrences represent the major cause for clinical deterioration and deaths, with a median survival of 12÷15 months following diagnosis [9]. The diffuse neutron beam could result suitable for the GBM tumour bed and local recurrences in the (superficial and deep) bed margins, with a certain selectivity of the cancer (vs. normal) cells thanks to the different RBE values. And, if clinically validated, nIORT® could be foreseen in multimodal treatment strategies, integrating surgery, chemotherapy and immunotherapy.

Acronyms

CNG Compact Neutron Generator

DD Deuterium-Deuterium

GBM Glioblastoma multiforme

HDPE High-Density Polyethylene

IORT Intra-Operative Radiation Therapy

LET Linear Energy Transfer

RBE Relative Biological Effectiveness

TC TheranostiCentre Srl company

Acknowledgment

This work was realised thanks to the precious help of Dr. Paolo Ferrari (ENEA) and the strict collaboration with Dr. Maurizio Martellini, scientific director of the Theranosti Centre Srl (TC, Milano) that financed the design and construction of the two CNG prototypes.

The computing resources and the related technical support used for this work have been provided by CRESCO / ENEAGRID High Performance Computing infrastructure (funded by ENEA and by Italian and European research programmes) and its staff, see *http://www.cresco.enea.it* for information.

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THE SELENE PROJECT: THE ITALIAN CONTRIBUTION TO A LUNAR BASE

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Abstract

In the last decade, the strategic importance of the Moon has been rediscovered internationally, both as ideal starting point for future missions to other planets or satellites in the solar system, and as environment that offers unique opportunities to increase knowledge of space. In this context, there is a growing interest worldwide in energy solutions based on surface nuclear reactors (SNRs), being a technology that allows high power densities.

The SELENE project was born in this scenario as an Italian contribution to a lunar base through the devise of a surface infrastructure, the Moon Energy Hub, specifically dedicated to storage and transmission of electrical and/or thermal energy produced in-situ by SNRs.

This project, co-funded by the Italian Space Agency and driven by a consortium composed of ENEA, the Polytechnic University of Milan and Thales Alenia Space Italia, has the overall objective of studying innovative technological solutions for the management of an SNR-based energy infrastructure for human and robotic activities that meets the criteria of operability, reliability and compactness suitable for applications in the lunar environment.

ENEA, in addition to being the project coordinator, is directly responsible for the following tasks: the design of the SNR; the analysis of safety, nuclear licensing and environmental impact related to the operation and decommissioning of the nuclear plant; and, finally, the design of the wireless energy transmission and reception systems.

This paper presents the status of the ENER division's activities related to the SNR design one year after its start.

Keywords: *Surface Nuclear Reactor; Moon Energy Hub.*

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Introduction

In the last decade, the strategic importance of the Moon has been rediscovered internationally. The Moon, in fact, is an ideal starting point for future missions to other planets or satellites in the solar system, as well as an environment that offers unique opportunities to increase knowledge of space.

In this renewed space race, the Italian Space Agency (ASI), like other national agencies, has decided to develop its own contribution to a lunar base. To this end, a call for proposals has been launched by ASI to finance three complementary projects: two dedicated to planning the scientific experiments to be carried out on the Moon and one aimed at producing the energy needed to make these experiments possible.

The SELENE (Sistema Energetico Lunare con L'Energia NuclearE) project, among the winners of this call, is focused on the concept of Moon Energy Hub (MEnH), responsible for the production, storage and transmission of electrical/thermal energy with one, or more, surface nuclear reactors at its center. The choice of the SNR technology stems from the fact that, due to its higher power density compared to other energy technologies such as solar cells or radioisotope power systems, it represents a viable solution for providing energy for high power activities close to or at medium distance from the basecamp.

The main objective of SELENE (see Fig.1) is to advance the design status and technology readiness of the Moon Energy Hub in order to pave the way for the further industrialization. This requires innovative technological solutions for the management of a SNR-based energy infrastructure for human and robotic activities that meets the criteria of operability, reliability and compactness suitable for applications in the lunar environment.

The project was conceived by a consortium composed of ENEA, in the role of project coordinator thanks to its experience in the nuclear reactor field, the Polytechnic University of Milan thanks to its inter-disciplinary experience for the Hub integration and, Thales Alenia Space Italia, as a strategic partner able to combine industrial vision and strong heritage in the space sector. The tasks for which ENEA is directly responsible are the following:

- the design of the SNR (except for the heat dissipation system);
- the preliminary evaluation of safety and licensing aspects of the SNR;
- the analysis of strategies for the decommissioning of the SNR;
- the analysis of the feasibility and performance of energy transmission and reception system using wireless methods (e.g., laser).

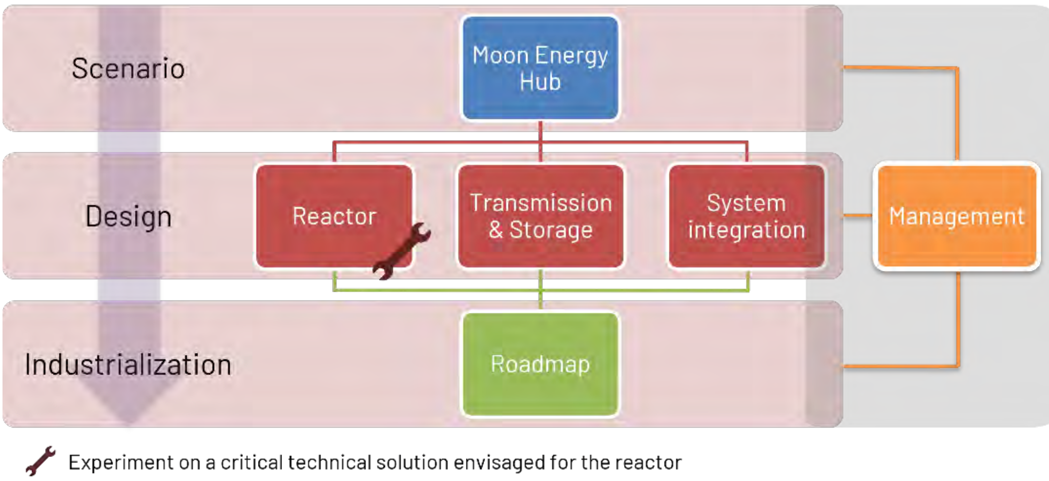


Fig.6: High-level structure of the SELENE project.

In particular, as regards the ENER division, the following activities should be noted:

- the definition of the high-level design requirements for the SNR consistent with the reference mission;
- the design of the SNR both in terms of the power production and of the thermal-to-electric conversion system;
- the realization of an experimental proof of concept for a low-TRL reference solution;
- the definition of the safety approach and path-to-licensing involving all life phases of the system (including launching);
- the analysis of the capacity of the Italian (primarily) and international supply chain to procure the main components required by the Moon Energy Hub.

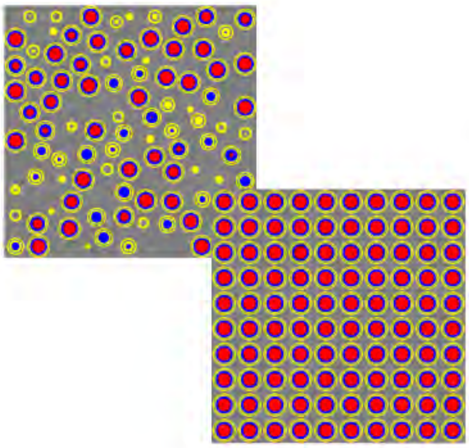
The status of the activities related to the SNR design carried out by the ENER division one year after its start will be depicted in the following.

ENER activities on SNR core design

The objective of the core design is to determine a geometric and material configuration capable of providing a given power for a given period of time, with size and mass constraints due to the launchers and landers. In this project, evaluating the opportunity window, it was decided for an electrical power of 100 kW for 10-year time, with a total mass lower than 8 tons. Although the design activities are closely dependent on each other and proceed in parallel, the traditional distinction between neutronics and thermal-hydraulics has been adopted in the following. It should be noted that this activity was able to take advantage of the previous feasibility study in which several options were compared and some suggestions were provided in terms of fuel type (TRISO particles), enrichment in ^{235}U (<20%, HALEU) and cooling system (heat pipes) [1].

Neutronics

The neutronic activities and analyses lean on the Monte Carlo code OpenMC. The focus has been directed on specific issue regarding the TRISO modelling to assess the type of effect that can be expected introducing computational alternative routes. Heterogeneous modelling (explicit particles with fixed kernel position) and two different homogeneous modelling (Volume-Weighted Homogenization and Reactivity-equivalent Physical Transformation) were compared with the reference solution represented by randomly placed fuel particles in the medium. Simulation issues regarding lattice cell edge intersection were investigated before moving forward and introducing a moderator. In fact, since HALEU fuel is employed, a moderator is crucial to increase the fission event probability and contain the reactor dimensions. Zirconium Hydride was considered the valuable candidate.



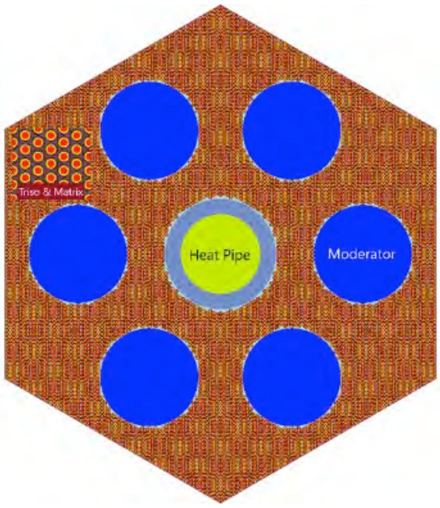


Fig.2: Heterogeneous Stochastic and Ordered models (on the left) and Prototypical Fuel Assembly (on the right).

At this point, a preliminary configuration of fuel assembly was designed adding a heat pipe sized according to thermal-hydraulic considerations (see Fig.2). Insights into the fuel-to-moderation ratio were subsequently derived, leading to an optimization of the configuration. Near future steps are the calculations of criticality, power distribution and reactivity coefficients of both fuel and moderator in order to reach at a first reference core layout configuration.

Thermal-hydraulics

Suitable thermal working conditions must be guaranteed for the SNR system. Heat removal from the core, heat transfer to the conversion system and downstream heat rejection system are the main aspects to consider.

Reference technology to carry out these tasks are heat pipes, that are passive devices that exploit evaporation and condensation of a working fluid to transfer heat from a hot sink, known as the evaporator, to a cold sink, called the condenser (see Fig.3).

The work done so far focused on:

- understanding the physics of heat pipes operation, characterizing their limits according to materials and structure composition of the main parts of which they are composed [2].
- selection of suitable working fluid and structural materials for both intrinsic characteristics (thermophysical properties, neutronics features and system compatibility) and usability for thermal constraints on system conversion requirements.

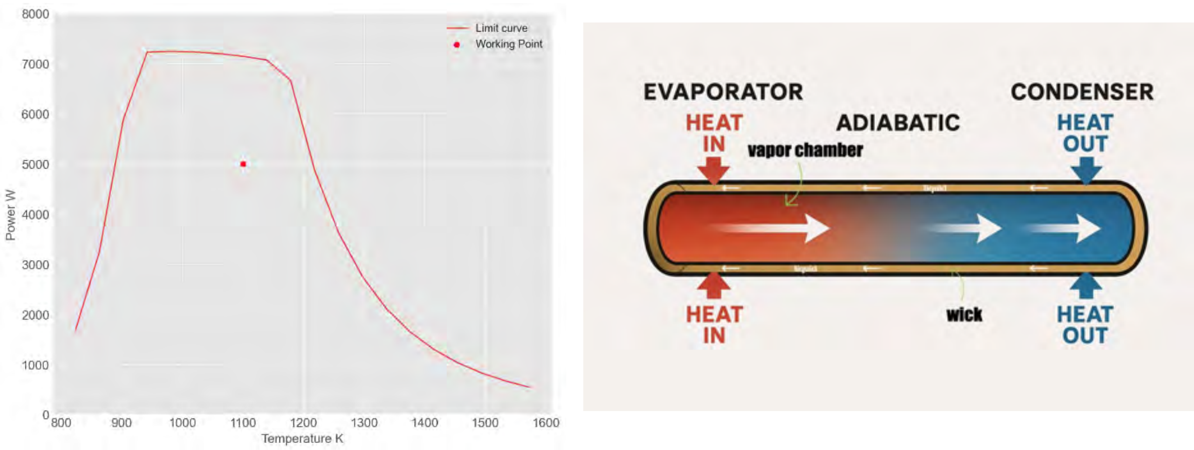


Fig.3: Heat Pipe working limit curve (on the left) and Heat Pipe operating principles (on the right).

Based on previous considerations, the selected working fluid is Sodium while viable structural materials could be Niobium alloys, superalloys or SiC. A preliminary geometry and corresponding performance limit curves of heat pipes have been evaluated and the resulting operating curve for the reference heat pipe can be seen in Fig.3. This design is characterized by a heat pipe able to transfer 5 kW at a nominal working condition of 1100 K.

The ongoing work is now focused on detailed CFD heat pipe modeling using OpenFoam [3]. This analysis will later be extended to the assembly and core levels, downstream suitable V&V of heat pipe model based on literature review of available data and future experiments to be conducted at ENEA's Brasimone Research Center using a dedicated test rig.

ENER activities on SNR system conversion

Previous analysis led to identify Stirling-cycle converters as the baseline technology for the temperature and power ranges. Among other converter types, Closed-Cycle Gas Turbines (CCGT) and Alkali-Metal Thermo-Electric Converters (AMTEC) were opted as backup solutions, primarily due to their limitations in terms of scalability/modularity and low maturity, respectively, when compared to Stirling technologies.

Further work will focus on the definition of a robust and flexible conversion architecture tailoring the system requirements. Furthermore, an optimization of Stirling converters units will be carried out to improve efficiency at high-temperature range regimes and reduce the converters' mass. The optimization will include material selection, specific components' geometrical parameters (i.e. heat exchangers), and system layout.

Previous analysis also highlighted that the mass reduction objective shall consider the mass penalties introduced by the heat rejection system, as the size (hence the mass) associated with this is significantly dependent on its working temperature. As shown in Fig.4, this parameter has a significant influence both on the converters' efficiency (hence the mass associated with the conversion system to achieve the nominal power output) and on the surface area required to radiate the excess heat to the outer space.

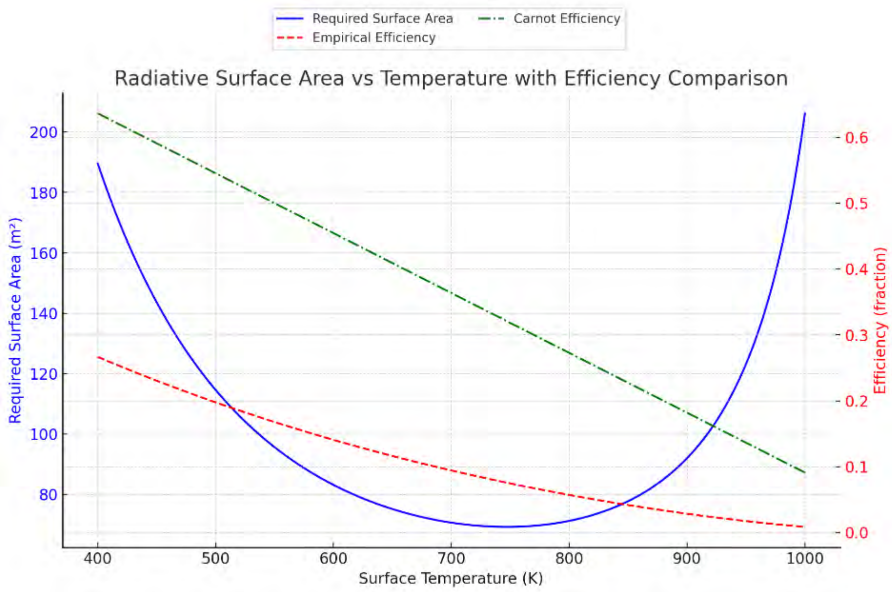


Fig.4: Converters' efficiency and surface area required to radiate the excess heat for Pout = 100 kWe. The red dashed curve represents the estimated conversion efficiency by extrapolation of experimental data for a GPU-3 type, Free-Piston Stirling Engine model.

Conclusions

The SELENE project is currently focusing on the design of the main components of the Hub spanning the SNR, energy storage and transmission systems with the ENER division firmly focused on the SNR. In parallel, engagement with relevant stakeholders is underway to expand the outreaches of the project in preparation for the final phase to be completed by June 2027. Indeed, as one of the main outcomes together with the conceptual design of the Hub and its deployment strategy, a programmatic roadmap to further propel the Italian industrial landscape in the race for the Moon will be developed including research and development activities still needed, regulatory gaps and supply chains considerations.

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DESIGN METHODS FOR BEAM INTERCEPTING DEVICES FOR PARTICLE ACCELERATORS

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Abstract

In engineering applications, Heavy Liquid Metals (HLMs) are used across various fields for the transport of momentum and energy. Another application involves their use as target materials for the interaction with particle beams and as damper to safely dispose of high energy beams. In this context, HLMs offer several potential advantages, thanks to their capacity to absorb high power density depositions. These properties make them promising candidates for Beam Intercepting Devices (BIDs) providing an alternative to solid materials. In nuclear science and high-energy physics, liquid metal BIDs are already explored in projects such as the MYRRHA reactor and physics experiments like ISOLDE. More recently, they are considered for proposed particles accelerators as the International Muon Collider Collaboration (IMCC) and the Future Circular Collider (FCC). This paper presents the engineering and numerical methods used in the development of these technologies, with a particular focus on BIDs for particle accelerators. Key physical phenomena are discussed such as heat deposition, shock wave generation, and magnetohydrodynamics (MHD) effects in multi-material systems using HLMs in argon-filled environments. The engineering challenges are addressed with the collaboration framework between CERN and ENEA, aiming at conceptual design of liquid metals' BIDs for the FCC and the Muon Collider. The FCC bremsstrahlung dump is designed to absorb a high-energy photon beam, delivering up to 500 kW in continuous operation. Meanwhile, the Muon Collider target is required to produce muons from fast proton beam pulses, delivering in the target 2-4 MW of thermal power on average.

Keywords: *Beam Intercepting Devices; Heavy Liquids Metals; CFD; Particle Accelerators; Shock waves.*

Introduction

Beam Intercepting Devices are fundamental components in particle accelerators, designed to manage the interaction with high-energy charged particle beams, either through spallation processes, target, or direct absorption, dumper. Currently, solid materials such as graphite and tungsten are the most used solutions. However, the continuous increase in beam energy required by the next generation of

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accelerators introduces significant challenges in terms of power density management and resistance to beam-induced damage. In this context, the use of heavy liquid metals is gaining increasing attention. These materials, due to their high density, enable more efficient pion production and, at the same time, exhibit thermo-hydraulic properties that make them well suited to dissipate large amounts of thermal energy. As a result, new technologies are currently under development that involve the use of liquid lead for the construction of targets and beam dumps [1][2][3]. In the past, experiments have also been carried out with liquid mercury, including those related to the Neutrino Factory and ISOLDE [4][5][6]. These experiments, conducted with high-energy beams, revealed complex issues, including shock wave generation, splashing, and the mechanical issues on the key components. The reflection of shock waves within the liquid metal can lead to cavitation, while the presence of strong magnetic fields (up to 20 T) induces electromagnetic and magnetohydrodynamic (MHD) effects [2]. Finally, a deep understanding of these phenomena is not only relevant for accelerator physics but also plays a fundamental role in the field of nuclear engineering, particularly in the context of Accelerator Driven Systems (ADS), as well as in the safety assessment of systems containing liquid metals under extreme operating conditions. In this paper we are presenting two design models under evaluation now in ENEA, the target for the IMCC end the dumper for the FCCee. These two design models help us to understand the challenges behind those technologies.

Methodology and Research Activity

Target: IMCC case

A shock wave study for the IMCC target is here presented. For studying the behaviour of the liquid target a CFD simulations are carried out to understand the magnitude of the shock wave and velocity. At this stage the only physical phenomena studied are the shock wave. This is due because the shock wave phenomena are the major phenomena that could damage the structure of the component, instead the MHD would be a drag for so would decrease the shock wave amplitude and velocity. The deposition in the fluid is modelled as a high-pressure region at the initial time. The pressure within the volume is estimated using the following Mie-Grüneisen equation of state. Using the value of the thermal source, the overpressure required to initialize the problem is obtained as,

$$p = p_{ref} + \frac{\gamma}{V}\left(E - E_{ref}\right),$$

where p, is the pressure, E, the energy source, γ the Mie-Grüneisen coefficient. The fluid motion is solved using the Volume of Fluid (VOF) method, assuming an inviscid three-phase mixture of liquid lead, lead vapour, and argon. To avoid negative pressure the Singhal cavitation model is used to allow the phase change for liquid lead to vapour of liquid lead at 10 Pa. The compressibility of lead is modelled using the Tait equation of state (EOS), which is widely used in UNDerwater EXplosion (UNDEX) simulations [8]. Since the Tait equation is independent of temperature the energy equation is not solved. Argon and lead vapour are modelled with constant physical properties, derived using the ideal gas law at reference pressure and temperature. The physical model is solved in ANSYS FLUENT, where the chosen equation of state (EOS) for lead is implemented via a User-Defined Function (UDF).

Dumper: FCCee case

For the FCC-ee, a liquid lead beam dump has been proposed by ENEA and is currently under evaluation in collaboration with CERN for integration and source term deposition analysis. The conceptual design is based on the characteristics of the power deposition profile, which exhibits a peak at the centre (0,0) in the transverse (x-y) plane and decreases with distance from the origin, following an approximately Gaussian distribution, figure 1. The figure 2 illustrates a possible design for the dumper, using a Lead Bottom Stream Target (LBST) configuration for the hight energy beam (multi material zone, MM-zone) and a pipe for the low energy beam (single material zone, SM-zone). Using Fourier's law, it is possible to calculate the temperature distribution within the steel plate to determine the threshold of the internal heat source below which the beam energy deposition does not pose a risk in terms of thermo-mechanical stress. The solution is given by:

$$T(x) = -\frac{q_g}{2k}x^2 + \frac{q_g L}{h}\left(1 + \frac{Lh}{2k}\right) + T_f;\ T(0) - T(L) = \frac{q_g L^2}{2k}$$

where q_g is the heat source, L the length of the steel palate, h and k are the heat transfer coefficient for convection and conduction respectively.

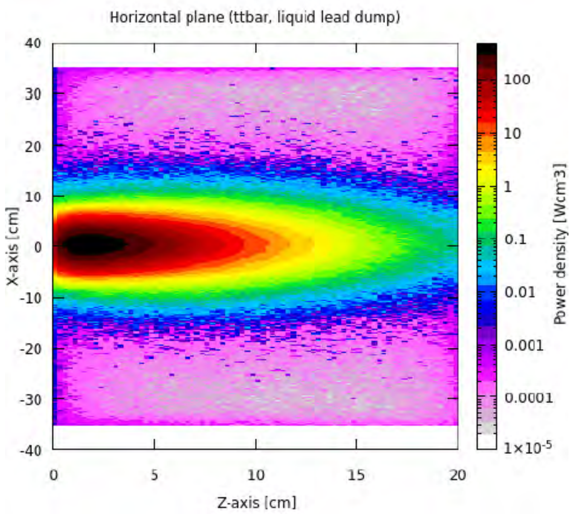

Fig.1: Power deposition into the liquid lead target- FLUKA simulation by CERN 2024.

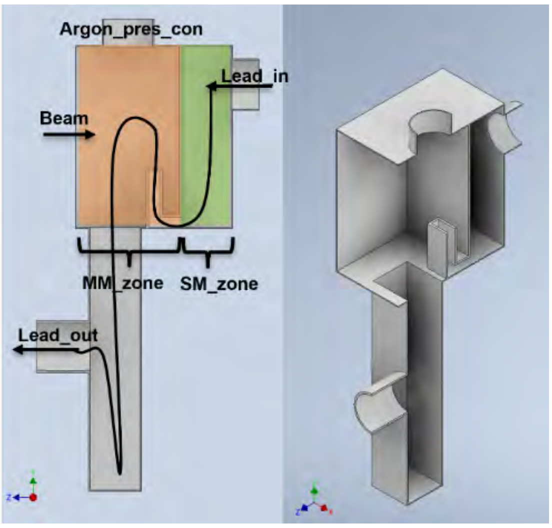

Fig. 2: Proposal design for the FCCee dump.

By comparing the temperature distribution in the plate, we define a reasonable threshold for the volumetric heat source, which requires coverage of an area of approximately 0.2 m × 0.05 m (D × L). To address this the LBST was designed to cover the required area. This configuration was chosen because this bottom stream due to the gravity naturally increases its cross-sectional area toward the centre. Using Bernoulli's equation, the required mass flow rate was estimated to be 180 kg/s. Based on these values, a ANSYS FLUENT VOF simulations were performed to assess the stability of the vertical jet.

Results

Target: IMCC case

The computational domains used are 2D configurations designed to model possible layouts of BIDs. To limit the propagation of shock waves to the surrounding structures, an argon buffer is interposed between the interaction zone and the structure. The simulations were carried out using a configuration that we call Lead Bottom Stream Target configuration. A PISO scheme and a time step of 1e–8s were used for the numerical setup. The figure 4 illustrates the BCs and simulation results for a beam interaction area with a diameter of 10mm and an overpressure of 5GPa, resulting after a thermal deposition of 10e7 J/m3. The figure 5 shows the behaviour of pressure and velocity over time at various points within the fluid domain.

Dumper: FCCee case

The VOF simulation for FCCee case is done to understand the stability of the jet and the effective cover of the hight region beam intercepting aria. A PISO scheme, an adaptive mesh and an adaptive time step are used. With these setups we achieve the stability of the jet as figure 6 shows.

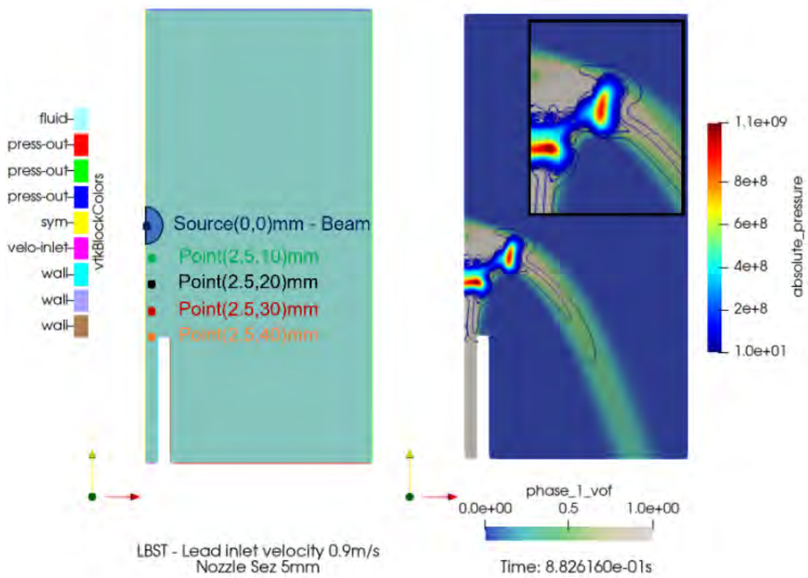

Fig.4: BCs (left) and contour of lead volume fraction with pressure isolines indicating the shock-wave region (right).

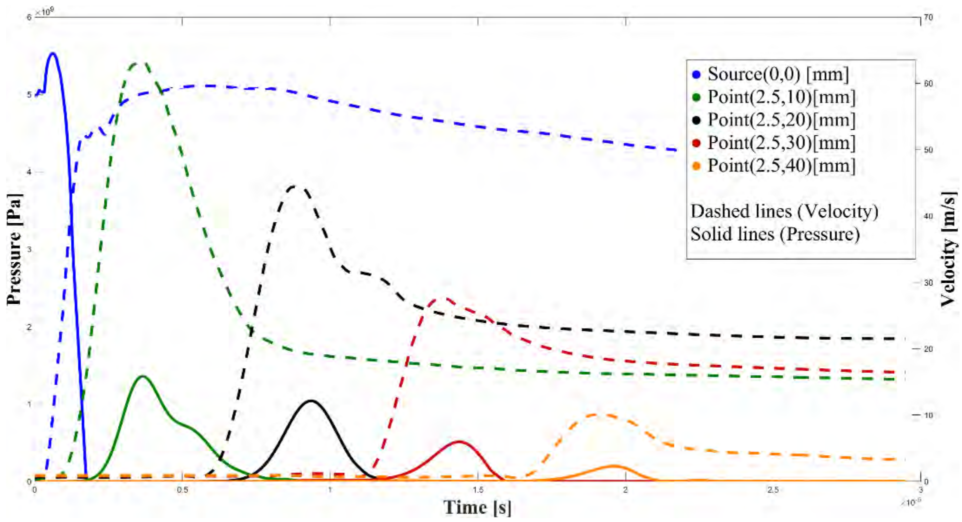

Fig.5: Pressure (indicating shock-wave passage) and velocity profiles over time at selected probe locations; time origin (t = 0) marks the onset of energy deposition after the flow reached steady state (0.8826s).

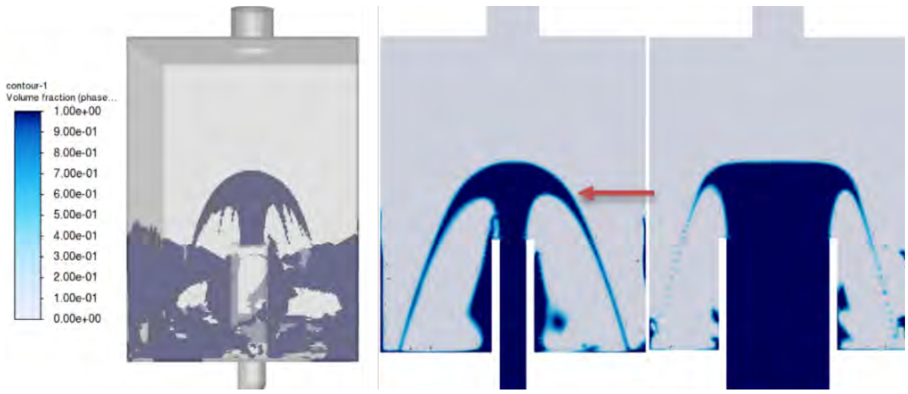

Fig.6: Volume of fraction of liquid lead for the LBST configuration.

Conclusions

Target: IMCC case

This preliminary study investigates the propagation of shock waves within BIDs caused by high-power beams. The results highlight that the shock wave period is on the order of 1e–5s and show the magnitude of the major physical variables such as pressure and velocity. They also reveal additional physical phenomena that require further investigation, such as cavitation in heavy liquid metals. Future analyses will focus on implementing a temperature-dependent cavitation model and comparing different EOS for heavy liquid metals.

Dumper: FCCee case

The FCCee CFD study highlights the possibility to use a LBST for the FCCee dumper, to reduce the mass flow rate, using a hybrid solution with a multi material zone and a single material zone. We conclude this option with 180 kg/s is really promising and it is an improvement of the baseline solution that is considering a five meters slope of liquid lead with 350 kg/s of mass flow rate. Future investigations are planned to better integrate the FLUKA results with this geometry.

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IMPROVING INITIALIZATION OF ASTEC'S THERMAL-HYDRAULIC SOLVER WITH MACHINE LEARNING-BASED METHODS FOR ENHANCED CONVERGENCE IN SEVERE ACCIDENT SIMULATIONS

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Abstract

Severe Accident (SA) simulation codes like ASTEC (Accident Source Term Evaluation Code) are essential for predicting nuclear reactor behavior under SA conditions. However, their high computational demands, particularly in thermal-hydraulic simulations (e.g., the CESAR module), can constrain their effectiveness. A significant portion of CESAR's computational load arises from solving non-linear Partial Differential Equations (PDEs) at each timestep using Newton-Raphson iteration. While Newton-Raphson convergence depends on initial guesses close to the final solution, current ASTEC implementation relies solely on previous converged states, without predictive insights.

This paper introduces a hybrid approach enhancing CESAR's Newton-Raphson solver through Machine Learning (ML) models, which offer refined initial guesses. Drawing on recent advancements, this approach explores ML-based surrogate models to learn intricate, non-linear relationships within transient conditions, aiming to reduce iterations needed for convergence and potentially allow longer timestep intervals without sacrificing accuracy. The surrogate model choice remains adaptable, balancing predictive accuracy and computational efficiency within CESAR's frequent initialization routines.

Preliminary results show that ML-augmented approaches can reduce ASTEC's computation time, suggesting promising implications for broader thermal-hydraulic simulation applications in nuclear safety assessments. Future work may involve developing surrogate model ensembles, complemented by classifiers that dynamically select optimal initialization based on reactor state and accident phase, optimizing solver performance across varying scenarios.

Keywords: *Severe Accidents, ASTEC/CESAR, Newton-Raphson, Machine Learning, Hybrid Solver*

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Introduction

Severe Accident (SA) codes are essential for nuclear-safety work: they simulate accident progression, test mitigation actions, and deliver source terms for emergency plans. Europe's reference tool is ASTEC (Accident Source Term Evaluation Code) [[1]], but its detailed physics makes large campaigns slow and expensive. The Horizon-Euratom project ASSAS [[1]] tackles this by embedding Machine Learning (ML) aids in ASTEC. One line of work speeds up CESAR's Newton-Raphson solver. Although quadratic in theory, Newton's convergence depends on a good initial guess; using the previous time-step can fail during fast transients, forcing costly time-step reductions. Hybrid "AI-assisted" solvers offer a remedy: ML models supply better initial fields, cutting iterations without loosening convergence criteria. Examples include Fourier Neural Operators that trimmed Newton steps by ~40 % in nonlinear PDEs [[2]], Newton-informed neural operators handling multiple solutions in one training, and reinforcement-learning schemes that adapt solver parameters on the fly. Other supervised hybrids have accelerated CO_2-storage flow models [[3]] and general ODE/PDE solvers [[4]], consistently lowering CPU time while preserving accuracy.

This study gives first results of such an approach for CESAR: an Artificial Intelligence (AI) model proposes smarter initial guesses, reducing iterations and runtime yet still matching reference solutions. Early gains are promising, though further work is needed to reach production-level speedups.

ASTEC – CESAR

ASTEC is a computational code designed to simulate SA scenarios in nuclear reactors. Its modular architecture enables the modeling of diverse physical phenomena, and each module operates independently, interacting through a centralized database (ODESSA) that ensures consistency.

The thermo-hydraulic core is CESAR, which solves the conservation equations of mass, momentum, and energy for liquid and gas phases in the primary and secondary circuits.

An implicit scheme advances the solution: ICARE, devoted to core degradation (oxidation, melting, fission-product release), receives thermo-hydraulic fields from CESAR. It returns decay power and geometry changes that alter coolant flow and predicts core state with previous flow fields. CESAR converges through finer internal micro-steps up to the synchronization time considering the new ICARE solutions; finally, ICARE corrects its estimates with updated results, closing the loop.

Numerical stability is controlled by a common macro-step and adaptive micro-steps for tightly coupled modules. The resulting nonlinear system is solved with Newton–Raphson: at each iteration the unknown increment is obtained by inverting the Jacobian until the residual norm drops below the threshold. If iterations hit the limit without converging, the time-step is halved and the process restarts.

After N successive failures the simulation stops, flagging instability; conversely, if convergence is reached in few iterations, the step is enlarged to reduce CPU cost. Fig.1 summarizes the link between iteration count and time-step.

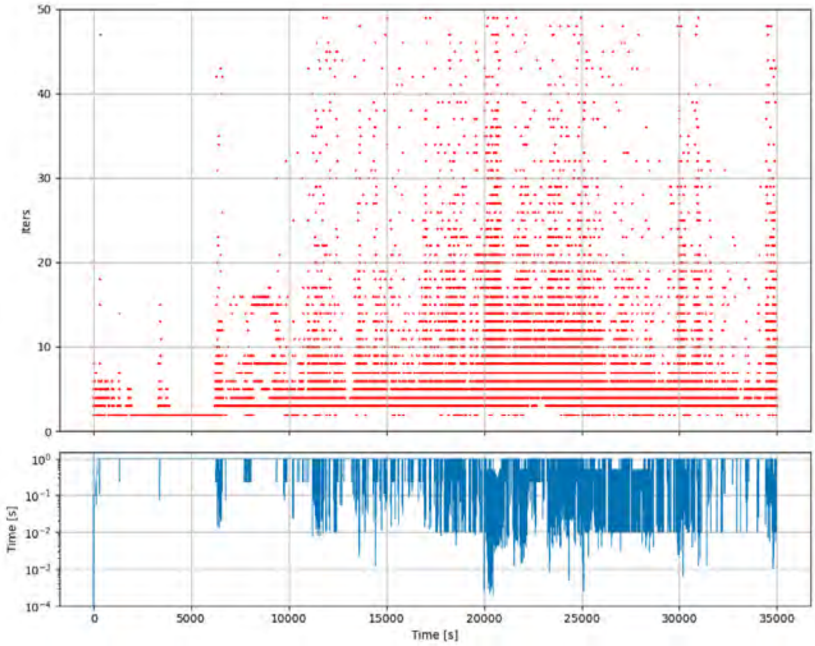

Fig.1: Relationship between iteration count and timestep adaptation.

The latest release of ASTEC (astecV3.1.2) lets users extract and tweak initial guesses for the balance equations, offering fine control on convergence.

Reference input deck PWR1300 LIKE and database

The study relies on the PWR1300-LIKE input deck released by IRSN within the ASSAS project [[5]], featuring two scenarios: Loss of Coolant Accident (LOCA) and Stationary Blackout (SBO); this preliminary analysis focuses on LOCA only. Each run records all internal CESAR iterations in binary format and aggregates them in a single HDF5 file: the outcome is a set of 2234 features (timestamp included) over several steps defined by the vessel rupture or three days.

Variables preserve ASTEC's native ordering: control volumes with partial pressures, void fractions, and temperatures; junctions with phase velocities; pump rotational speeds; inner and outer wall temperatures. The spatial arrangement mirrors the case discretization and may be reorganized in future to cluster contiguous volumes for better readability.

Methodology

Hybrid Newton-Raphson techniques enhanced by AI have so far been tailored to single-equation CFD cases with fixed boundary conditions and uniform grids. ASTEC/CESAR, in contrast, must resolve tightly coupled, nonlinear mass-, momentum- and energy-balance equations characteristic of nuclear thermal-hydraulic transients, so existing approaches transfer poorly.

Two factors render ML-based initialization difficult: (i) highly inter-dependent state variables whose prediction errors quickly destabilize the solver, and (ii) adaptive, irregular time-steps that erase periodic structure and force the model to forecast across a window rather than a single instant.

Conventional RNN or CNN schemes presume regular sampling; resampling the data can lead to a large information loss. Therefore, successful reconstruction and subsequent forecasting must embed time information directly. Continuous-time stochastic models (CARMA), continuous-discrete Kalman filters and Gaussian Processes (GP) can cope with arbitrary sampling but scale poorly for nonlinear, large-volume data. Modern deep-learning variants solve this by incorporating elapsed-time into their architecture (GRU-D [[6]], Phased-LSTM), parameterizing latent dynamics through differential equations (Neural ODEs [[7]], NCDEs [[8]]), or weighting observations via temporal attention (e.g., mTAN [[9]]). These ideas underpin our solution.

Proposed Architecture

Our model (Fig.2) fuses continuous-time embeddings with a multi-head Probabilistic Attention (Prob-Attention) layer. Raw timestamps are mapped into a high-dimensional space, after which Prob-Attention samples the most informative key-value pairs, preserving long-range temporal dependencies at moderate cost. An encoder of alternating attention and convolution blocks captures global and local dynamics; residual links and layer normalization stabilize gradients. The mirrored decoder reconstructs physical variables without any preliminary interpolation to validate the hidden state, then enables forecasting for Newton-Raphson initialization.

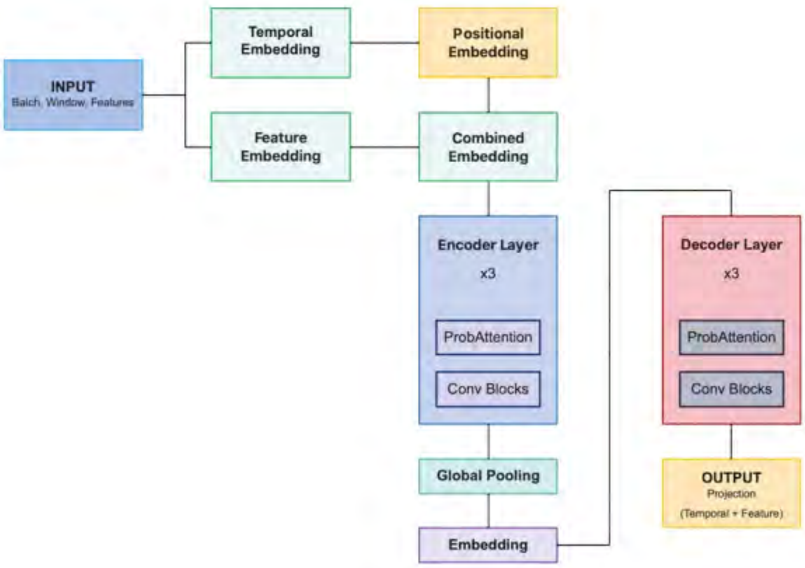

Fig.2: Model Architecture.

Data Processing and Training Setup

The pipeline discards duplicates and missing points, rescales all features to [0, 1], and transforms each timestamp into fifteen engineered temporal descriptors—normalized progress, multi-scale interval deltas, Fourier-based cycles—so that the network learns, rather than assumes, temporal structure. Sliding windows with fixed history and stride preserve full multivariate information essential for physically consistent initial guesses.

Training (PyTorch on CPU, GPU forthcoming) minimizes a composite loss: Mean Absolute Error plus penalties for violating physical bounds (pressures ≤ 22.1 MPa, void fractions 10^{-5}–0.999 999, gas/surface temperatures 273.15–5273 K, liquid 273.15–646.85 K). Bayesian tuning selects learning rate, batch and epochs; dropout, early stopping and cross-validation suppress over-fitting. Performance is finally assessed via MAE, RMSE and MAPE before integrating the predictor into ASTEC.

Results and Discussion

The transformer-based autoencoder demonstrates promising reconstruction capabilities for ASTEC - CESAR thermal-hydraulic dynamics, achieving MSE = 0.0242, MAE = 0.1446, R^2 = 0.841, and correlation = 0.9998 across 2,234 coupled variables.

Variable-specific analysis reveals distinct reconstruction quality patterns. Temperature fields exhibit the best performance (MSE ~0.021-0.023, correlation >0.994) due to their spatial-temporal continuity and higher thermal inertia. Flow variables achieve excellent reconstruction (MSE 0.0175-0.0197, correlation >0.90) despite turbulent two-phase flow complexity. Pressure variables show moderate performance (MSE ~0.031-0.033, correlation ≥0.997), with higher errors attributed to sharp gradients during valve operations and rapid depressurization. Void fractions demonstrate reasonable reconstruction (MSE ~0.033, correlation 0.992), while pump rotational speeds exhibit the poorest performance (MSE 0.0298, correlation 0.891) due to discrete operational states and mechanical-thermal coupling.

The model successfully captures complex nonlinear relationships and preserves physical correlations. However, it exhibits systematic bias with a compression effect in mid-range values (0.3-0.7), temporal smoothing that attenuates sharp transients during rapid changes, though it maintains physical constraint compliance with no unphysical negative values observed.

Fig.3 demonstrates the model's temporal reconstruction capabilities, showing excellent tracking of gradual variations with minor deviations during rapid transients around t=15,000s. The model exhibits characteristic smoothing behavior typical of autoencoder architectures when dealing with high-frequency components, which may benefit Newton-Raphson solver initialization by avoiding numerical instabilities from discontinuous initial conditions.

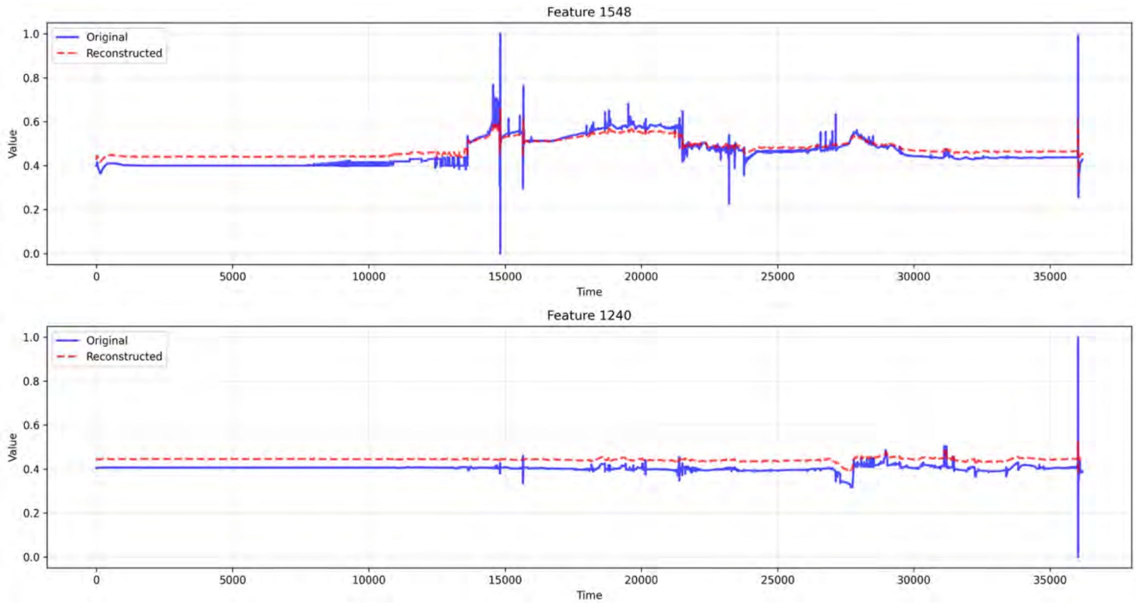

Fig.3: Comparison Original vs Reconstructed – preliminary optimization.

Key challenges requiring future work include systematic estimation bias toward mean predictions, lack of explicit spatial information in dataset structure, missing boundary conditions from other ASTEC modules (ICARE), and limited effectiveness during extreme transient conditions. The results establish a foundation for ML-enhanced Newton-Raphson initialization while highlighting the need for architectural refinements and dataset enhancement to achieve practical deployment in safety-critical applications.

Conclusions

The transformer-based architecture achieves promising reconstruction performance (MSE = 0.0242, correlation = 0.9998) across 2,234 variables but exhibits systematic bias and struggles with extreme transients.

Key limitations include lack of spatial organization in the dataset and missing boundary conditions from other ASTEC modules. Future work requires dataset enhancement, physics-informed architectures, and ensemble models. Despite challenges, this hybrid ML approach demonstrates potential for accelerating nuclear safety simulations and establishing foundations for real-time severe accident prediction.

Acknowledgements

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SESSION 4

EXPERIMENTAL CAMPAIGNS AND FACILITY UPDATES

Chairperson: *C. Carrelli*

Session 4 was aimed at providing an overview of recent experimental activities and infrastructure developments supporting nuclear science and technology. The session highlighted the central role of experimental facilities as enablers for advanced research, validation of numerical tools, and qualification of technologies relevant to current and future nuclear systems.

Several contributions focused on cutting-edge experimental campaigns carried out in heavy liquid metal facilities, addressing key phenomena relevant to reactor design and safety. Experimental investigations on flow-induced vibrations in fuel pin bundles and on thermal-hydraulic behaviour under representative operating conditions were presented, demonstrating the importance of dedicated test sections, advanced instrumentation, and tailored data acquisition and post-processing techniques. These activities contribute to improving the understanding of complex fluid–structure interactions and heat transfer mechanisms in lead-cooled systems, providing valuable input for design and validation purposes.

Other presentations addressed materials performance under aggressive operating environments. Experimental campaigns devoted to corrosion testing of advanced coatings and structural materials in liquid lead were illustrated, highlighting the role of controlled exposure conditions, detailed post-test characterization, and systematic comparison of different mitigation strategies. These studies represent a key element in supporting the deployment of lead-cooled reactor technologies and in identifying solutions to enhance materials durability at high temperatures.

The session also included contributions related to facility upgrades and revamping activities, illustrating how existing experimental infrastructures are continuously adapted to meet evolving research needs. The presented experiences emphasized the challenges associated with facility modification, commissioning, and calibration, as well as the importance of post-test analysis to improve data quality, reduce uncertainties, and fully exploit experimental results. These activities were shown to be essential for ensuring the reliability and relevance of experimental campaigns.

Finally, the importance of large-scale international experimental infrastructures for nuclear data production was highlighted. Measurements of neutron-induced reactions carried out at high-resolution facilities were presented as a fundamental contribution to the improvement of nuclear data libraries, with direct impact on reactor physics, safety analysis, and advanced system modelling. The strong interaction between experimental measurements, sensitivity studies, and application-driven requirements was emphasized as a key driver for prioritizing and guiding experimental efforts.

Overall, the session demonstrated how coordinated experimental campaigns, continuous facility development, and advanced measurement techniques constitute a cornerstone of nuclear research. The presented activities clearly reflected the capability of ENEA and its partners to operate and evolve state-of-the-art experimental infrastructures, delivering high-quality data and supporting innovation across a wide range of nuclear applications.

FLOW INDUCED VIBRATION (FIV) EXPERIMENT IN THE HELENA FACILITY

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Abstract

The activity described is the experimental setup and results for the Flow Induced Vibration (FIV) in a fuel pin bundle test section (FPS) with flowing lead. The test section is a 37-pin mock-up representative of the GEN-IV ALFRED Fuel Assembly (FA) and is manufactured to investigate the turbulence induced vibrations in the pins. The pins are instrumented with longitudinal strain gauges (SG) and with 3 SG per monitoring point, the displacement signal will be recovered in terms of amplitude and frequencies. To achieve this goal, a complex algorithm is implemented in the data acquisition control system of the test section.

The test section was installed in the HELENA Heavy Liquid Metal loop at the ENEA Brasimone R.C., in which a mechanical pump for lead circulation is present. The test matrix is proposed with mass flow rate varying from 10 to 50 kg/s and constant temperature of 450°C for all the tests. Briefly, the experimental procedure is presented to carry out the experimental campaign.

Experiments were carried out, and results showed RMS maximum oscillation of 14 microns at the maximum flow rate of 50 kg/s. The maximum oscillation was around 40 microns at the maximum flow rate in the most excited pin. Vibrations are negligible for flow rate lower than 35 kg/s. A frequency analysis of the experimental data will also be presented.

Conclusions are that oscillations in the fuel pin bundle are of low entity and are not dangerous at nominal flow rates in the ALFRED FA.

Keywords: *Flow Induced Vibration (FIV); Heavy Liquid Metal (HLM); Fuel Pin Bundle Simulator (FPS); Experiment.*

Introduction

The paper describes the Flow Induced Vibration experiment done in the HELENA facility at the ENEA Brasimone R.C.. For the realization of the experiment, a new test section, the Flow Induced Vibration Fuel Pin Simulator (FIVFPS), was manufactured and mounted in the HELENA facility, and the experiment to assess turbulence induced vibration in the fuel pin bundle of the ALFRED Fuel Assembly in flowing lead was carried out. For a description of the HELENA facility and of the test section for FIV

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and for the experimental procedure please refer to [1]. For the modal analysis of the elementary pin in lead please refer to [2].

HELENA Facility

HELENA is the acronym for Heavy Liquid metal Experimental loop for Nuclear Applications and is an AISI316L rectangular loop to test fluid dynamics, heat transfer and other phenomena. The working fluid is pure liquid lead. The loop can be operated with liquid lead in a temperature range between 380°C and 550°C.

The facility consists of a primary rectangular loop with flowing lead and a secondary loop with pressurized water. The primary loop consists of 2.5″ S40 AISI316L pipes with connecting flanges WN 2.5″ 300lb ASME B16.5, a heat exchanger and an expansion tank. A centrifugal online mechanical pump guarantees a mass flow rate up to 80 kg/s. For the purpose of the experiment, the secondary loop will not be used, and the primary lead will maintain approximately a constant temperature during a single test, being the heating induced by the mechanical pump negligible in a few minutes, which is the typical duration of the test. A thermocouple will measure the temperature of the lead before entering the test section. The mass flow rate is measured with a Venturi flow meter in the loop coupled with two bubble tubes (with argon) and a differential pressure transducer in the gas. From error analysis, it emerges that the error of the measurement of the mass flow rate is less than 2% in all cases.

FIVFPS Test Section

An experimental test section (FIVFPS) was designed to study the Flow Induced Vibrations (FIV) in the ALFRED FA with flowing lead. With respect to the 127 pins of the ALFRED FA, a 37 pin FA mock-up is considered for the test section. The test section is designed to be relevant for the ALFRED FA with 1:1 scale in the streamwise vertical direction. The whole test section, including the pins, is manufactured in AISI316L. The test section is equipped with coupling flanges to connect it to the HELENA facility. The duck neck on the top of the test section allows the instrumentation signals to go out of the system.

The single pin must be representative of the ALFRED pin, with a total length of 1710 mm and an active length of 810 mm. The external diameter of the pin is D = 10.5 mm with a cladding thickness δ = 0.6 mm and an internal diameter D_i = 9.3 mm. The bundle is characterized by a hexagonal lattice with pitch P=13.6 mm.

To respect the mechanical vibration behavior of the pins of the ALFRED FA it is necessary to reproduce the weight distribution along the pins. For this reason, in the active region, the pins are filled with holed pellets of tungsten carbide to reproduce exactly the weight of the fuel in the ALFRED FA pin (510 g in an active length of 810 mm, i.e., 629 g/m). With a density of tungsten carbide of around 14700 kg/m^3, to maintain the fuel weight, it is necessary to adopt holed pellets with an external diameter of 9 mm and an internal diameter of 5.13 mm. The single pin reproduces exactly one of the ALFRED reactor, and in particular, all other internal structures and components. Based on a literature review, longitudinal strain gauges (SG) were identified as the preferred solution for monitoring vibrations. The choice is justified by the fact that other proven methods like fiber-optic sensors (see for example [3]) have strong temperature limits.

To capture the first vibration modes, the SG are installed in the mid-plane between the two grids (first mode) and at ¾ of the distance (second mode). Assuming the origin of the *z* vertical axis at the pin bases in the bottom grid, with a distance between grids of 1700 mm, SG are positioned at $z = 660$ mm (plane A) and $z = 1258$ mm (Plane B).

Regarding the azimuthal position of the SG, *Fig. 2* reports the position of the SG in the planes A and B. The figure reports the bundle seen from the top with the numbered pins and the azimuthal angle reference system to identify the SG. It should be noticed that in each instrumented pin, three strain gauges (called 'triplet' in the following) at 120° are positioned in the same monitoring point. This feature allows the reconstruction of the amplitude of oscillation through an appropriate post-processing of the three measurements. All the details of the post-processing algorithm are documented in [4], including the nomenclature for the different quantities involved. Moreover, the Fast Fourier Transform analysis of the derived signal allows for fixing the dominant frequency of the oscillation.

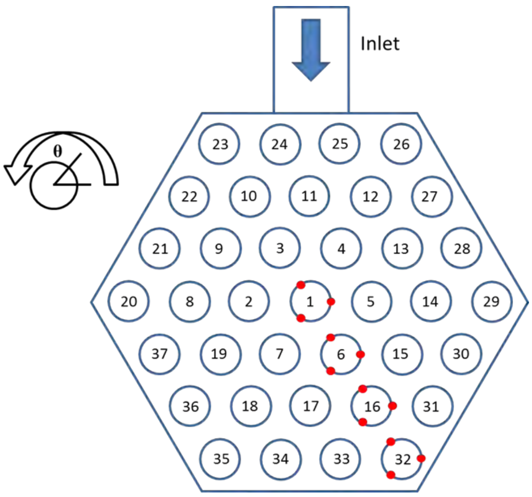

Fig. 2: Schematic layout of the Fuel Pin Bundle FIVFPS seen from the top in the two monitoring planes.

Experimental Tests

The experimental tests have been carried out by the lead flow in the experimental test section as stated in the previous sections. It was clearly evidenced for example in [4] that the main cause of vibration is turbulence in the flow.

To limit the noise of the SG signals, the tests were finally conducted at all flow rates with the heating cables of the test section switched off. Tests are repeated three times for repeatability. 21 tests were conducted (3 slots of 7 tests at different flow rates each). The slots are the following:

- Slot 2 (with heating cables switched off, 7 tests at increasing flow rates according to the test matrix reported in *Table 3*): duration 2 minutes each test.
- Slot 3 (with heating cables switched off, 7 tests at increasing flow rates according to the test matrix reported in *Table 3*): duration 2 minutes each test; made for repeatability.
- Slot 4 (with heating cables switched off, 7 tests at increasing flow rates according to the test matrix reported in *Table 3*): duration 2 minutes each test; made for repeatability.

Table 3 provides the exact flow rate in the different tests of slots 2, 3 and 4. The mass flow rate is identical in the same test of the different slots because the pump speed is fixed.

Table 3: Mass flow rates and pump rotation speed for the different tests in slots 2, 3 and 4.

Test number (x is referred to the slot = 2,3,4)	x.1	x.2	x.3	x.4	x.5	x.6	x.7
Mass flow rate [kg/s]	12,97	20,29	29,79	35,36	40,82	44,89	49,04
Pump rotation speed [RPM]	200	300	420	500	580	640	700

Raw data treatment

The signals for the SG have a typical drift caused by thermal and electrical effects present in the SG. To remove this effect, very low frequencies have been cut by a filter to eliminate low-frequency disturbances and drift. Moreover, very high frequencies have been cut off as well to improve the signal/noise ratio and the subsequent analysis of the data. It is well known from previous analysis [2] that typical physical frequencies of the involved phenomena are of the order of some Hz. Keeping in account all these considerations, a bandpass filter was applied to all the raw signals with a bandpass of 0.25–30 Hz.

Analysis of the amplitude of the deflection h

Applying the rigorous algorithm to the results documented in [4] and the look-up tables for the deflection h as a function of the curvature radii Ra and Rb, for the 9 cases 2.5, 2.6, 2.7, 3.5, 3.6, 3.7, 4.5, 4.6, 4.7, results show a maximum amplitude of 40 micron for test 2.7, pin16.

The maximum value can be considered as an upper limit of the deflections. More interesting from an engineering point of view is the RMS values shown in *Fig. 3* for slot 4.

The figure clearly shows that the deflection increases linearly with the mass flow rate, and this is physically reasonable and intuitive. The concordant deflections (where points A and B deflect from the same side) are always higher than the discordant and must be considered the reference curves. RMS oscillations of less than 15 microns are revealed.

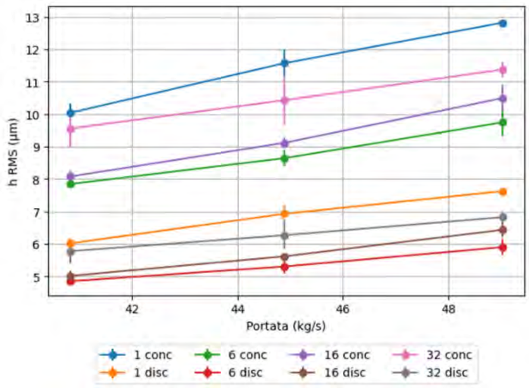

Fig. 3: RMS of the amplitude of the deflection for the different flow rates for slot 4.

Analysis of the frequencies

A frequency analysis was carried out with the SG signals. Results are shown in *Fig.* 4 for test 7 (about 50 kg/s).

The analysis shows that some frequency clusters emerge:

- Two peaks are present at 12 and 24 Hz about corresponding to 1X and 2X of the pump rotation speed.
- A frequency of 50Hz is tied to the effect of the heating cables of the loop, and this component is filtered with the bandpass filter for deflection analysis.
- A peak is present around 4 Hz corresponding to the measured typical frequency of oscillation; it was verified that this peak is present at all the significant flow rates.
- A low frequency peak is present, and it is probably due to noise.

From analysis, it emerges that the vibration physical peak is at 4.25 Hz and it is independent of the flow rate. This value is different from the eigenfrequency of the pin in lead computed in [2], equal to 8.76 Hz. The reason for this disagreement must still be investigated. Similar analysis can be repeated for all the significant cases in the different slots with similar results.

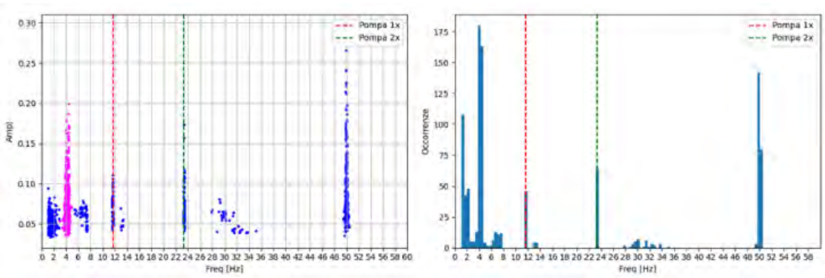

Fig. 4: Amplitude of the frequency peaks for test 4.7 in terms of amplitude of the peak (left) and occurrences (right).

Conclusions

The present paper describes the Flow Induced Vibration experiment performed in the lead HELENA facility at the ENEA Brasimone Research Centre. The experiment was carried out by manufacturing the FIVFPS test section instrumented with strain gauges (SG) to measure vibration frequency and amplitude. The test section is a 37-pin mock-up of the ALFRED FA and reproduces the top and bottom grid of the FA. All ranks of pins were monitored in the two planes A and B, and three SG will be installed for each monitoring point to capture the relevant modes of oscillation. An algorithm allows for deriving vibration amplitude from the measured voltage of SG [4]. This allows for reconstructing the amplitude signals of the vibrations. Results were in terms of frequency spectrum and vibration amplitude. A test matrix was carried out for the experimental campaign with mass flow rate ranging from 10 to 50 kg/s, reproducing the lead velocity in the ALFRED FA.

The main facts emerging from the data analysis are summarized in the following:

- The instrumentation and the post-processing algorithm were able to capture the amplitude of deflection, although these are of the order of tens of microns.
- The amplitude of the deflections h grows linearly with the flow rate and becomes significant for flow rates higher than 40 kg/s; this fact was interpreted as a predominance of the noise for lower flow rates.
- The maximum deflection measured with the rigorous approach is around 40 microns.
- The RMS deflection measured at the maximum flow rate is around 15 microns; therefore, it is very small from an engineering point of view.
- There is a substantial agreement between rigorous and approximate approaches for the computing algorithm.
- Frequency analysis showed a measured frequency of 4.25 Hz; this value is about half of the eigenfrequency in lead coming from numerical pre-test analysis [2], and the reason for that must be understood.

The experimental study offers a good overview of the pin vibration behavior and can be used for engineering evaluations and to assess computational methods.

Acknowledgment

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LEAD CORROSION PERFORMANCE OF Al AND Cr BASED PACK CEMENTATION COATINGS ON 316Ti WITHOUT PRE-OXIDATION TREATMENT

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Abstract

Lead-cooled Fast Reactors (LFRs) are considered among the most promising Generation IV reactor designs due to their high thermal efficiency and the inherent safety provided by using lead as a coolant. However, the aggressivity of the heavy metal in contact with structural materials at temperatures higher than 480°C represents a big challenge to improve the efficiency of this technology. Along the different strategies to mitigate the lead corrosion, the coatings deposited by the pack cementation technique have shown promising results. Preliminary studies on this technique have demonstrated the ability of this coating, together with adequate pre-oxidation treatments, to form a protective Al_2O_3 surface layer to reduce the mass loss and lead penetration on the austenitic steel.

In this framework, both Al and Cr based pack cementation coatings on 316Ti austenitic steel were exposed in static lead in the *new*cleo CAPSULE facility to evaluate their corrosion performance without pre-oxidizing treatment. The lead bath conditions were set at 650°C for 500h with an oxygen concentration of 10^{-7} wt. %. The samples underwent characterization by SEM/EDS to study the microstructure and elemental composition before and after exposure. Results show a general corrosion limited at the outer diffusion layer for the Al-based sample, and localized corrosion sometimes involving all the coating thickness for the Cr-based one. Further studies are underway to identify the most efficient pre-oxidation conditions to form protective surface oxides on pack cementation coatings.

Keywords: *Aluminum; Chromium; Pack cementation; Pb corrosion*

Introduction

The Lead-cooled Fast Reactors (LFR) are one of the most promising options of IV Generation reactors. The choice of liquid Pb as a coolant has significant advantages, such as the excellent shield effect for gamma radiation due to its high mass number, the absence of high chemical reactivity, avoiding explosive phenomena, and the possibility of natural circulation allowed by its high thermal expansion

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and low viscosity [1]. In addition, given its high boiling point (1737°C), Pb gives the reactor the ability to work at high operating temperatures, thereby increasing its energy efficiency.

However, corrosion of austenitic structural steels caused by contact with liquid Pb at elevated temperatures (>480°C) limits the possibility of improving reactor performance. Corrosion in Pb environments occurs mainly through two mechanisms: alloying elements dissolution, leading to Pb penetration in the material and resulting in superficial defects formation and embrittlement, and formation of surface oxides on the steel. To mitigate this phenomenon, in addition to innovative steels, various types of surface coatings are being investigated to improve corrosion resistance while maintaining the core properties of the material. Among them, the Al-based coatings deposited by the pack cementation technique have shown promising results [2]. Pack cementation is a diffusion deposition technique used to apply coatings to metal components, allowing the formation of adherent intermetallic layers of Al, Cr or Si. The resulting coatings exhibit excellent adhesion, thicknesses ranging from tens to hundreds of μm, and high uniformity, even on complex geometries. Preliminary studies on this technique have demonstrated the ability of this coating, together with adequate pre-oxidation treatments, to form a protective Al2O3 surface layer, which is expected to be even more effective than the as-deposited Al-based coating in reducing the mass loss and Pb penetration on the austenitic steel [3].

Methodology and Research Activity

In this framework, Al and Cr-based pack cementation coatings on 316Ti austenitic steel were exposed to static Pb to evaluate their as-deposited corrosion resistance at high temperature, without performing a pre-oxidation treatment. The Pb bath conditions were set at 650°C for 500h with an oxygen concentration of 10^{-7} wt. %. The experiment was performed in the *newcleo* CAPSULE facility (*Fig. 1*), composed of 18 static lead corrosion cells and corresponding storage vessels.

The samples are 40 x 10 x 3 mm plates with coating deposited all over the surface. Three samples were prepared for each coating type, Al and Cr, two of which were tested in Pb, while one was left virgin to assess the quality of the coating prior to the corrosion experiment. Chemical etching ($HCl:H_2O_2$) was performed on cross-sectional samples, to highlight the grain boundaries and microstructure of the coatings. To study the microstructure and elemental composition before and after Pb exposure, optical microscope (OM), scanning electron microscope (SEM) and microanalysis (EDS) were used, allowing high-resolution observation of the morphology and microstructure of steel and coating, and qualitative and semi-quantitative elemental chemical analysis.

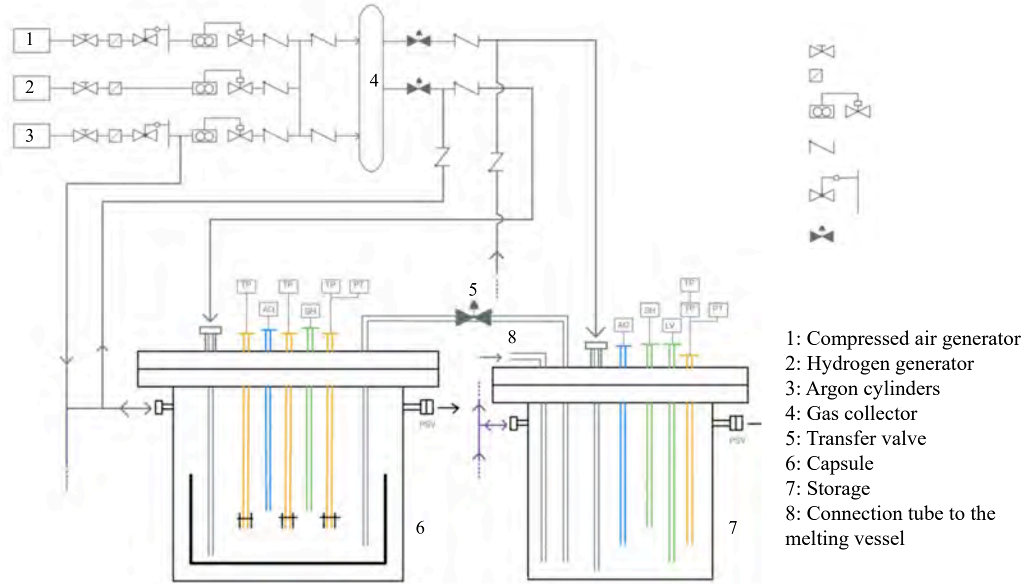

Fig. 5: Schematic drawing of a capsule-storage system of the newcleo CAPSULE facility

Fig. 2 reports the metallographic analysis, performed with an optical microscope at high magnification, on chemically etched cross-sectional samples after Pb exposure. In both cases, the bulk microstructure is fully austenitic, with no evidence of residual ferritic phases. Al-based coated sample (*Fig. 2-A*) clearly exhibits the presence of a two-zone surface coating. The outer region is characterized by grains with a morphology similar to that of the bulk austenite, but showing localized cracking, while the inner zone shows a columnar structure. Cr-based coated sample (*Fig. 2-B*) presents coarser grains, and the surface coating is barely visible under optical microscopy, suggesting either a thinner layer or low contrast with the substrate, which limits its observability with this technique.

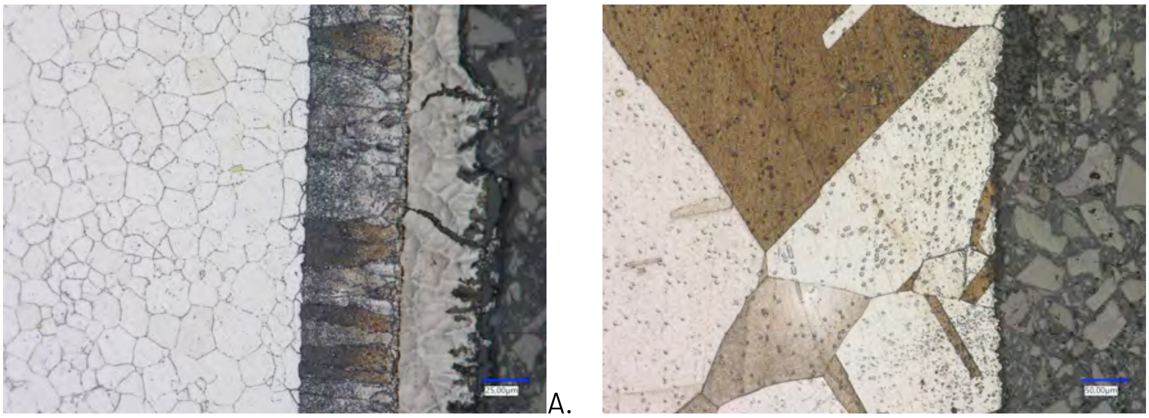
A. B.

Fig. 6: Optical microscope images highlighting the microstructure of the samples after Pb exposure: A. shows the Al-based coated and B. the Cr-based coated sample

The *Figure 3* cross-sectional SEM analysis, performed using backscattered electrons (BSE), provides more detailed insight into the microstructural and compositional variations across the coating regions of the two samples. In Al-based coated sample (*Fig. 3-A*), the outermost portion of the coating exhibits localized areas where material appears to be dissolved due to Pb penetration along grain boundaries. This hypothesis is supported by the presence of bright white spots, identified as entrapped Pb, and highlighted by arrows in the image. In the inner region of the coating, bright lines between the columnar grains suggest the presence of intermediate phases, although no signs of corrosion are observed in this area. Cr-based coated sample also reveals a bilayered coating structure, not visible through optical analysis, with an outer zone enriched in Cr and an inner diffusion zone. This coating appears to be intact in some regions (*Fig. 3-B*), while in others it is completely dissolved through its entire thickness (*Fig. 3-C*).

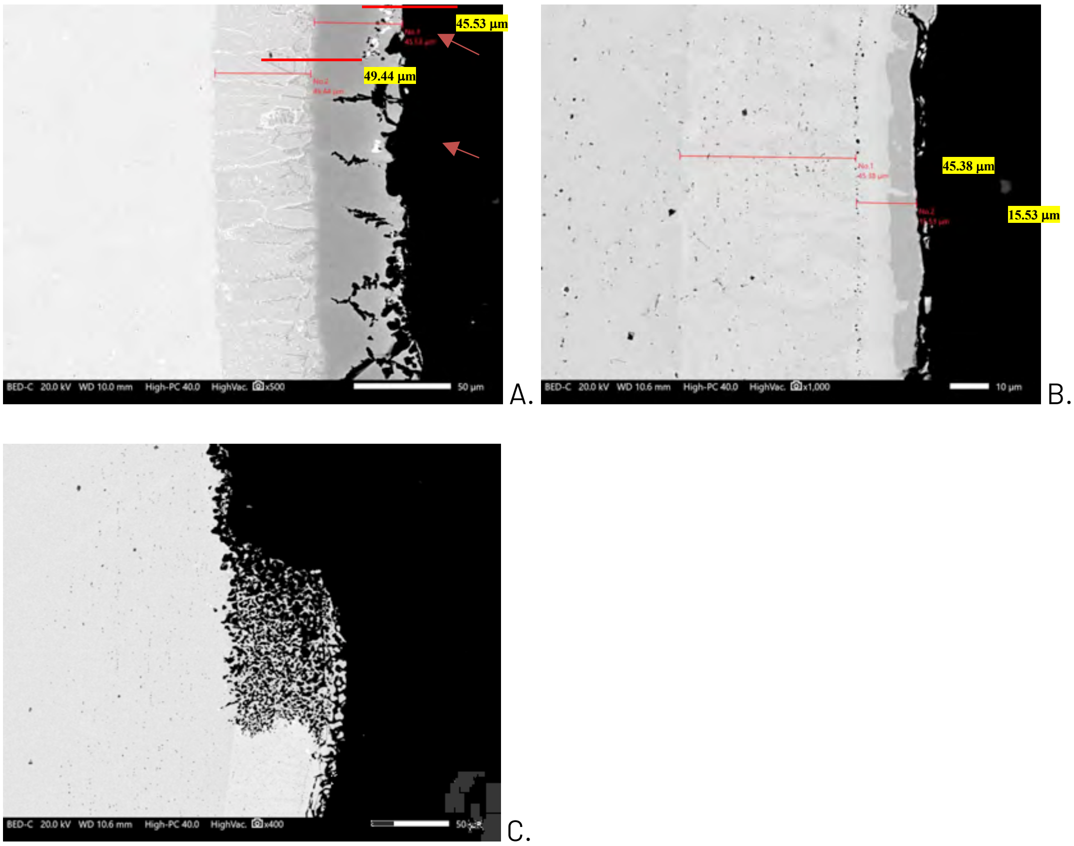

Fig. 7: SEM images of the samples taken with BSE after Pb exposure: A. shows the Al-based coated and B. and C. the Cr-based coated sample

The compositional mapping performed on the cross-section of the Al-based coated sample shown in *Fig. 4*, reveals a clear elemental stratification corresponding to the two regions of the coating. The outermost layer is enriched in Al, while the inner one is more Fe and Cr rich. There is a Ni concentration on the surface of the coating and on the interface between the two zones, indicating the possible presence of mixed phases. No surface oxide is detected though the map in this area.

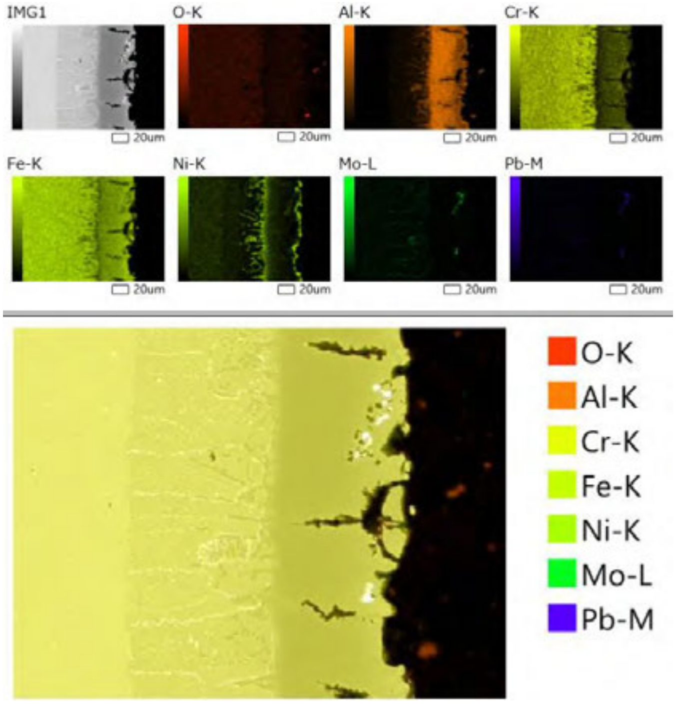

Fig. 8: Compositional map of the Al-based coated sample

Regarding the Cr-based coated sample, the EDS map in *Fig. 5* reveals the presence of a distinct stratified structure. In particular, a thinner surface layer with a higher Cr concentration is clearly visible, followed by a transition region with lower Cr content and increasing presence of substrate elements such as Fe and Ni. A higher concentration of Ni is observed at the interface between the coating and the substrate. In the observed area, the map shows the presence of oxygen in the outer Cr-rich region, suggesting possible surface oxidation.

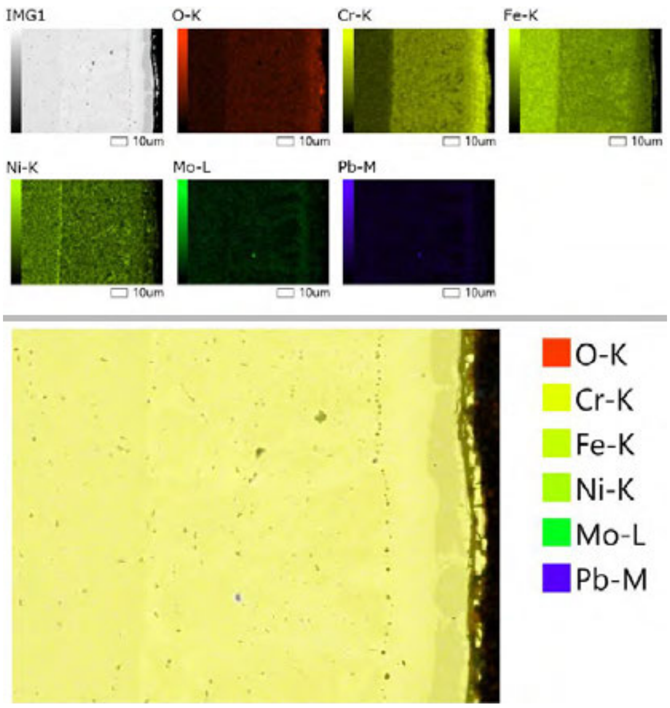

Fig. 9: Compositional map of the Cr-based coated sample

Conclusions

Overall, the two samples exhibit different corrosion behaviors: the first sample shows mild, generalized dissolution limited to the outermost coating layer, whereas the second sample undergoes severe, localized corrosion resulting in complete dissolution of the coating in certain areas. In both cases, the surface oxide formed on the coating surface is not sufficiently uniform and compact to completely avoid Pb corrosion.

Further studies are underway to identify the most efficient pre-oxidation conditions to form protective surface oxides on pack cementation coatings to improve their Pb corrosion resistance.

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NACIE-LHT REVAMPING AND EXPERIMENTAL TEST

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Abstract

The NACIE LHT facility, located at the ENEA Brasimone Research Centre, is the result of a revamping initiative carried out by *new*cleo. The primary coolant has been converted from Lead-Bismuth Eutectic (LBE) to pure lead. As part of the upgrade to the existing layout, a second heat exchanger was added, dedicated to investigating lead heat transfer under conditions representative of *new*cleo's steam generator (LHT test section); additionally, an electromagnetic pump was installed to replace the previous gas injection system.

An initial series of tests performed with the new configuration revealed opportunities for improving both the facility's operation and the quality of the collected data, prompting further post-processing activities. This paper presents an overview of the revamping process, with particular attention to the challenges involved in converting an LBE-based facility to one operating with pure lead. A description of the experimental conditions is provided, showcasing the capabilities of the upgraded system in terms of operating temperature and mass flow rate. Furthermore, the paper discusses key lessons learned from the experimental campaign and the outcomes of the data post-processing. The analysis demonstrates how post-test calibration and interpretation of raw data contributed to a deeper understanding of the thermal-hydraulic phenomena involved and significantly enhanced the overall quality and reliability of the results.

Keywords: *Calibration, Experiment, NACIE, Revamping.*

Facility description and experimental set-up

ENEA–*new*cleo partnership includes the utilization of selected existing ENEA facilities for *new*cleo's R&D program. Among these is the NACIE loop, which ENEA has used over the years in various experimental campaigns [1]. It has now been modified into the new NACIE-LHT configuration to pursue new objectives.

NACIE-LHT

NACIE-LHT (Fig.1) is named after its Lead Heat Transfer test section, which is designed to study *new*cleo's Steam Generator (SG). The goal is to identify an appropriate correlation between the Nusselt

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and Peclet numbers that accurately describes lead heat transfer in the SG geometrical configuration [2].

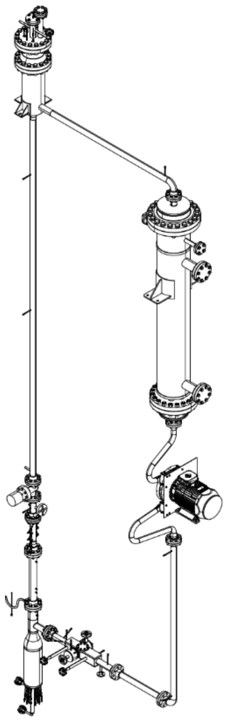

Fig.1: NACIE LHT loop layout.

The primary side of the loop consists of two horizontal pipes (2.5" in diameter), each approximately 2.4 meters long, and two vertical pipes with a total height of 7.7 meters. The modifications made during the facility revamping involve the following:

- the working fluid has been changed from Lead-Bismuth Eutectic (LBE) to pure lead;
- the bottom leg of the loop has been adapted to incorporate the LHT test section;
- an electromagnetic pump has been installed to allow forced circulation, replacing the previous gas injection system;
- an alternative secondary circuit has been added: an open loop operating under ambient conditions and connected to the LHT test section.

The remaining components of the loop have been maintained, including:

- the Fuel Pin Simulator (FPS) at the bottom of the riser;
- the Heat eXchanger (HX) in the upper part of the downcomer, and the pre-existing single-phase water secondary circuit, though it is currently disconnected;
- the expansion vessel located at the top of the riser.

Switching to pure lead as the working fluid introduces several critical operational challenges for the experimental campaigns:

- pure lead has a freezing point approximately 200 °C higher than that of LBE, requiring careful management during the operation of the LHT test section, particularly during the start-up of the secondary loop;
- to replicate reactor-like thermal conditions, the loop operates at higher temperatures, increasing the influence of heat losses on the experimental results, since no update of the insulation has been carried out;
- some instrumentation, such as pressure transducers, operates near its upper temperature limits.

The first experimental campaign of NACIE-LHT represents a key milestone for *new*cleo in developing expertise in operating lead-based facilities.

Section 2 provides a brief description of the test section and an overview of the test cases carried out in the first experimental campaign. Section 3 summarizes the key lessons learned from post-test data analysis and the subsequent calibration activities.

LHT test section

The design of the LHT test section (Fig.2) is based on the reference geometrical configuration of the *new*cleo LFR-AS-30 SG. The SG consists of a stack of spiral tubes arranged in a staggered layout. In this configuration, lead flows radially from the inner side of the SG to the outer side, passing through the tube bundle, with its velocity decreasing along the flow path.

This experimental campaign focuses on investigating convective heat transfer on the lead side. The LHT test section consists of a box through which a horizontal tube bundle passes. The geometrical parameters on the lead side are representative of those in the reactor's SG. The thickness of the water tubes has been increased to accommodate ambient temperature and pressure water entering the test section, minimizing the risk of thermal shock to the tubes.

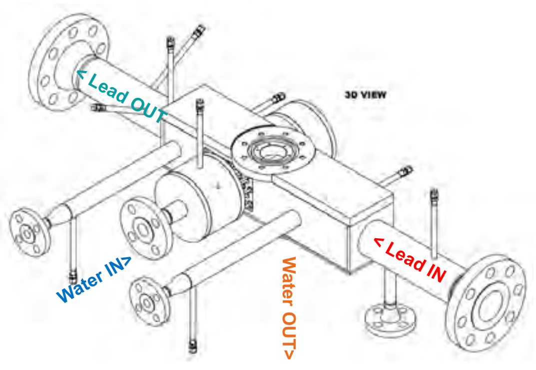

Fig.2: LHT test section.

First experimental campaign

The experimental cases tested during the first experimental campaign:

- cover the range of lead velocities expected during SG operation;
- cover the range of lead temperatures anticipated under typical SG operating conditions.

The minimum lead temperature must always remain above the freezing point of lead, even in stagnant or low-velocity zones between tubes. In these regions, particular attention must be paid to ensure the temperature remains above the melting threshold to avoid solidification.

Multiple thermocouples (TCs) have been installed within the test section to monitor temperature distribution and detect potential cold spots.

The experimental campaign consists of a series of steady-state tests conducted under constant water conditions, specifically fixed mass flow rate and inlet temperature, and varying lead mass flow rate and inlet temperature.

Results and post-test activities

The experimental results have been collected and processed to determine the operating conditions of the LHT test section, based on measurements taken throughout the loop. Local temperature measurements from thermocouples (TCs) within the test section box were also recorded, although they are not discussed in this work.

Experimental uncertainties were evaluated by considering the following sources of error:

- instrument error, as specified by the manufacturer;
- measurement chain error, determined through post-test evaluation;
- uncertainty due to signal fluctuations in the acquired data.

To estimate the uncertainty of derived quantities (u_f), primarily the Nusselt number, computed by the function $f(x_i)$, the following equation has been used [3]:

$$u_f = \sqrt{\sum_{1}^{n}\left(\frac{\delta f}{\delta x_i}u_{x_i}\right)^2}$$

The results obtained from the initial experimental campaign were deemed unsatisfactory due to significant power imbalance within the LHT test section and high values of uncertainty. To address these issues and enhance the quality of the data, a series of corrective actions and additional activities were undertaken. The following paragraph discusses the main quantities that were investigated as part of this effort.

Water mass flow rate and loop thermocouples

The Water Flow Meter (WFM) and the TCs installed in the loop were calibrated a posteriori to reduce uncertainties and improve the accuracy of the experimental measurements. The WFM operates using ultrasonic measurement technology and is installed upstream of the inlet water collector.

The calibration procedure involved filling a 1000-litre tank and comparing the ratio of the measured water weight to the total test duration against the mass flow rate recorded by the WFM. Time measurements were taken using a chronometer, and water mass was determined using a certified balance. Tests were performed at constant pump speed and valve opening, with three different valve positions, each repeated three times to ensure repeatability.

The results of the calibration tests showed that the WFM systematically underestimated the water flow rate (see Fig.3).

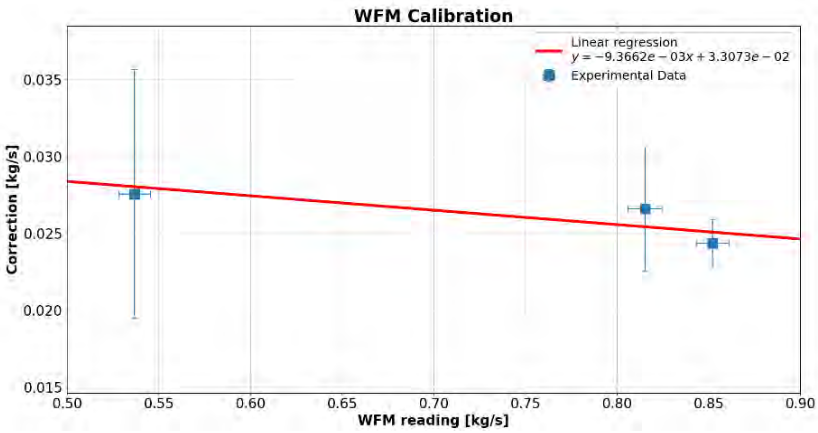

Fig.3: Water flow meter calibration curve.

The most relevant thermocouples (TCs) from both the lead and water loops, including those involved in the LHT test section heat balance, were removed and sent to a certified laboratory for calibration. Based on these results, a correction curve was derived for each TC to improve the accuracy of the temperature measurements.

Lead mass flow rate

In the primary loop, a Thermal Mass Flow Meter (TFM) is installed. By design, it consists of an inlet and an outlet thermo-resistance (TR), along with a heat source (up to 6 kW). The mass flow rate is determined from the heat balance across the TFM. This measurement concept, developed by ENEA, has proven reliable in the range of 0.5–5 kg/s, with the potential to operate at higher flow rates, albeit with reduced accuracy (Fig.4). Before the experimental campaign, the inlet TR failed. As a result, a thermocouple (TC) positioned upstream of the TFM was used in its place for the heat balance. However, the TC has a higher measurement uncertainty than the TR, which significantly impacted the accuracy of the mass flow rate estimation, especially at the higher flow rates observed during the tests (up to 13 kg/s).

To reduce the uncertainty, the mass flow rate was alternatively estimated based on the heat balance of the Fuel Pin Simulator (FPS), which operated at power levels between 70 and 85 kW. This approach involved evaluating heat losses associated with the power generation system by modelling the following contributions: in the cables between the power supply and the pin connectors; to air from the cold tail of the pin; from the FPS through its lateral surfaces; in the loop sections between the FPS and the TC locations used for the heat balance.

The final mass flow rate estimation incorporates all these factors and shows discrepancies of up to 10% when compared to the TFM measurements. As expected, this discrepancy decreases with decreasing lead mass flow rate.

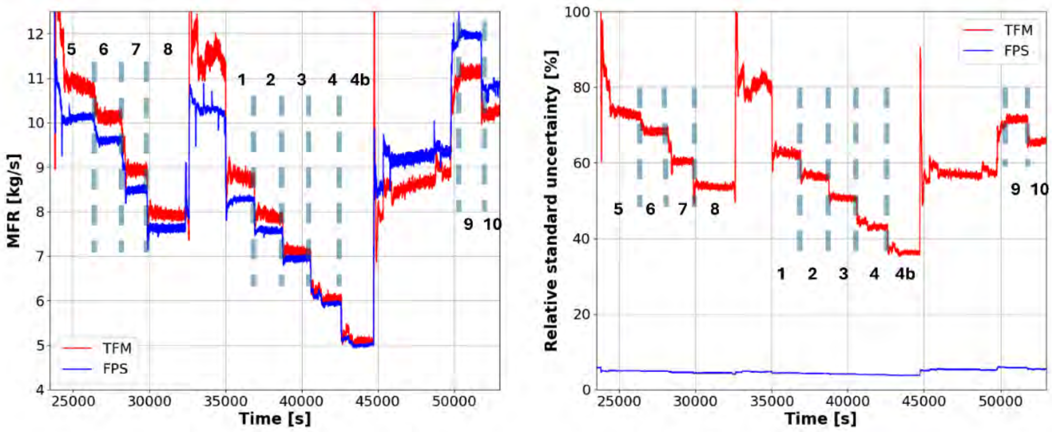

Fig.4: Lead flow rate during the test day. Numbered steady-state conditions are highlighted.

LHT power balance and Nusselt number accuracy estimation

The various activities led to a recalculation of the power balance within the LHT test section, which now also includes an estimation of heat losses obtained through Finite Element Method (FEM) simulations.

As a result, the measured power discrepancy between the heat exchanged on the lead side and on the water side has been reduced by nearly an order of magnitude (Fig.5).

This work has also significantly improved the uncertainty associated with the calculated Nusselt number. For instance, in TEST 4bis, the uncertainty decreased from 97% to 22%, representing a substantial enhancement in the reliability of the experimental results.

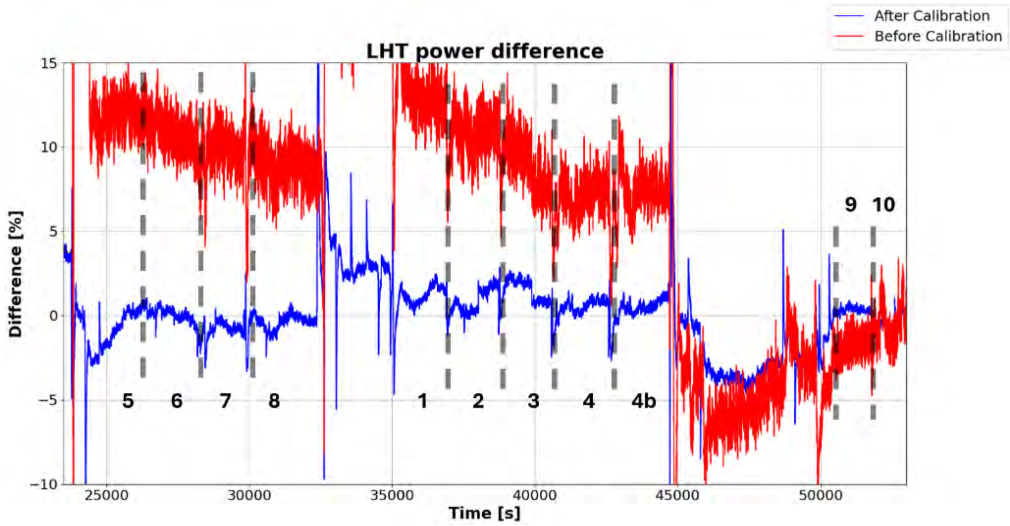

Fig.5: Power unbalance during the test day. Numbered steady-state conditions are highlighted.

Conclusions

The NACIE-LHT facility has been successfully modified to accommodate the LHT test section, specifically designed to study lead heat transfer in *new*cleo's steam generators. The activities on the first experimental campaign have led to significant improvements in data quality by addressing power imbalances and measurement uncertainties. Through post-test calibration and detailed modeling of the facility, discrepancies in the power balance were reduced by nearly an order of magnitude, and the uncertainty in the Nusselt number was significantly decreased, greatly enhancing the reliability of the experimental results. This campaign has provided valuable lessons for future experiments, contributing to the advancement of *new*cleo's lead-cooled reactor technology.

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MEASUREMENT OF ^{63,65}Cu NEUTRON CAPTURE CROSS SECTION AT THE N_TOF FACILITY

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Abstract

Neutron-induced reactions on copper are of great relevance for nuclear technologies, with additional interest in astrophysics concerning the s-process modeling in massive stars. Copper is a key functional material in the TAPIRO research reactor at ENEA, a facility central for fast reactor research and validation of nuclear data for Generation IV systems, such as Lead-cooled Fast Reactors (LFRs). Sensitivity and uncertainty studies on TAPIRO have underscored the impact of ^{63}Cu and ^{65}Cu cross sections on simulation accuracy, revealing inconsistencies in existing evaluations and a need for improved data to support advanced reactor development. To address these needs—primarily driven by nuclear technology requirements—the RAMEN initiative was launched within the Euratom-APRENDE Project, in partnership with the n_TOF collaboration at CERN. A measurement campaign targeting ^{63,65}Cu(n,γ) and ^{63,65}Cu(n,tot) cross sections is currently underway at n_TOF, a high-resolution time-of-flight facility that provides a broad neutron energy range (meV to GeV). The current campaign focuses on radiative capture and total cross sections measurements for both isotopes, conducted at the EAR1 experimental area. In parallel, feasibility studies are being carried out to prepare for future measurements of elastic and inelastic scattering channels, which are also of high priority and listed in the NEA High Priority Request List (HPRL). Preliminary results from the 2024 measurement campaign will be presented, marking a step toward the improvement of copper nuclear data.

Keywords: *Cu; Cross Sections; Fast Reactors; Nuclear Data; Sensitivity Analysis; TAPIRO*

The n_TOF facility

The neutron time-of-flight facility n_TOF at CERN is recognized as one of the leading infrastructures worldwide for high precision measurements of neutron-induced reaction cross sections. Data produced at n_TOF are essential for a wide range of applications, including dosimetry, medical applications, reactor safety, nuclear waste transmutation, fuel cycle studies, and the development of advanced reactors like Generation IV and fusion systems. In addition to its technological relevance, n_TOF also contributes to fundamental research through studies on nuclear level densities and processes relevant to stellar nucleosynthesis [1].

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The time-of-flight method

Neutron-induced reaction cross sections often exhibit resonance structures that are highly isotope-dependent and hardly predictable form theory alone. Consequently, accurate cross sections require excellent energy resolution across a broad energy range, spanning from thermal energies to hundreds of MeV. Time-of-flight (TOF) technique is among the most effective ones for such measurement. In this approach, neutrons produced in short pulses travel along a known flight path L before interacting with the target sample. The kinetic energy E_n of each neutron is inferred from its time of arrival t, using the relation:

$$E_n(t) = m_n c^2(\gamma - 1) \qquad (1)$$

Where $\gamma = (1 - v^2/c^2)^{-1/2}$ is the Lorentz factor, $v = L/t$: is the neutron velocity. The energy resolution is evaluated as

$$\frac{\Delta E_n}{E_n} = \gamma(\gamma+1)\frac{\Delta v}{v} = \gamma(\gamma+1)\sqrt{\left(\frac{\Delta t}{t}\right)^2 + \left(\frac{\Delta L}{L}\right)^2} \qquad (2)$$

Long flight paths can better resolve resonant structures in cross sections (large L and t) and by producing neutrons in the shortest possible time (Δt) and space (ΔL) intervals. For this reason, the n_TOF facility at CERN provides unique high-resolution measurements.

The neutron beam

n_TOF is a fixed-target experimental facility at CERN. Proton beam pulses are extracted each 1.2 s from the Proton Synchrotron (PS) and delivered to the neutron-producing target in bunches of 7×10^{12} protons with a time width of 7 ns, relevant for time-of-flight resolution.

Protons impinge on a lead target and many particles are produced, among which neutrons, γ-rays, protons, pions, fragment nuclei and neutrinos. The neutron beam is generated through this spallation process, and all neutrons are assumed to be emitted simultaneously. Prompt emission γ-rays cause saturation in the detectors of the experimental area.

To broaden the neutron energy spectrum to thermal energies, neutrons are partially moderated as they escape from the target using layers of water or borated water. This reduces neutron capture by hydrogen, suppressing the production of further γ-rays.

Along the neutron beamlines, charged particles are removed from the beam using “sweeping magnets”, while collimators and additional shielding elements define the beam aperture. The final beam diameter in both areas is set by a shaping collimator located just before the experimental area.

The experimental areas

The n_TOF facility operates with three experimental areas, each tailored to specific measurement requirements. The experimental area 1 (EAR1), located 185 m away from the spallation target, is optimized for high-resolution measurements, particularly in the resonance regions, allowing measurements up to GeV neutron energies.

The experimental area 2 (EAR2) is placed vertically 20 meters above the spallation target (the neutron flux has a factor of 30 times higher than EAR1): optimal when high statistics is essential. NEAR is the third experimental area, which is the closest to the target.

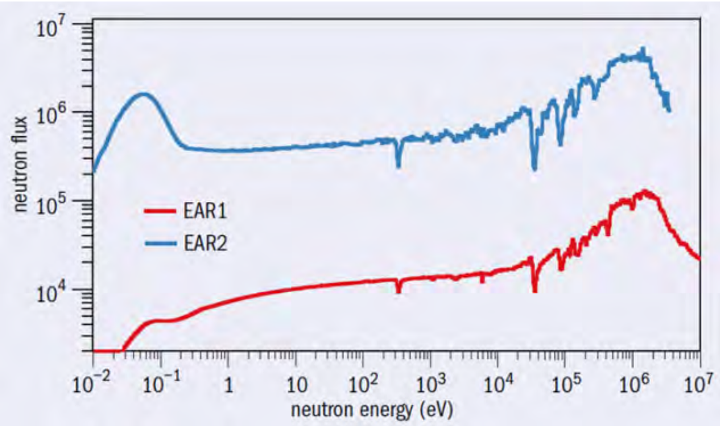

Fig.1: Neutron counts per proton bunch on spallation target function of neutron energy (EAR1 and EAR2). Reduction of thermal peak in EAR1 is due to the neutron capture in borated water.

Capture experiments at n_TOF

Detection systems are composed of both solid and liquid scintillators, primarily used for detecting γ-rays following (n, γ) reactions. Gaseous and solid-state detectors measure (n,charged particles) reactions instead. Fission reactions are monitored with gaseous detectors.

In a neutron capture reaction, an incoming neutron is absorbed by the target nucleus, forming a compound nucleus in an excited state. This excited nucleus typically de-excites to a lower energy state, often the ground state, by emitting a cascade of γ-rays. The pattern of this γ-ray cascade can vary with the energy of the incident neutron, particularly when transitioning between resonances in the cross section, as each resonance corresponds to a different excited state of the compound nucleus.

The Total Absorption technique

The Total Absorption technique is based on detecting the complete γ-ray cascade emitted following a neutron capture event. Detectors used for this purpose have high photopeak efficiency, fine segmentation, large solid angle coverage, good energy resolution, fast timing, and low sensitivity to neutrons.

The Total Absorption Calorimeter (TAC) consists of a 4π array of 40 BaF_2 crystals, providing high precision in cross section measurements. However, the TAC's relatively high neutron sensitivity and its slow detector response impose limits on the maximum neutron energy that can be reliably measured. These factors complicate measurements on materials with significant neutron scattering cross sections. While this detector is particularly suited for (n,γ) measurement of actinides, it is not the optimal choice for Cu(n,γ).

The TED technique

The TED technique detects capture events proportionally to the total energy of the emitted γ-ray cascade, making it independent of the de-excitation path. Achieving this proportionality is inherently challenging, as the compound nucleus can de-excitate through many possible decay routes. Thanks to their fast timing and low neutron sensitivity, C_6D_6 are the best choice for light and intermediate-mass elements such as copper. Their response requires a mathematical correction, known as the Pulse Height Weighting Technique (PHWT), to ensure proportionality between γ-ray energy and detection efficiency [2].

An array of 4 C_6D_6 detectors was employed at n_TOF for the capture measurements of ^{63,65}Cu.

Data analysis of capture measurements

The data reduction process begins with the raw digitized signals recorded by the C_6D_6 detectors and follows a sequence of steps to extract the capture yield, from which the capture cross section is derived via an R-Matrix analysis.

First, a pulse shape analysis is performed to identify and characterize individual detector signals, followed by the construction of energy spectra. In parallel, the response of the neutron beam monitors is analysed to characterize the neutron flux incident on the sample, its energy is determined from the measured TOF. A critical stage in the analysis is the background subtraction.

To account for the γ-ray detection efficiency and ensure proportionality to the total energy released in the capture process, the PHWT is applied. The resulting yield is then absolutely normalized using the Saturated Resonance Method [3], that relies on well-known resonance strengths. The neutron energy range over which the extracted yield is valid is obtained.

The Pulse Height Weighting Technique

A TED detector is required to exhibit low intrinsic efficiency, typically only one γ-ray from the capture cascade is detected at a time, and its efficiency must be proportional to the γ-ray energy, i.e., $\varepsilon(E) =$

$k \cdot E$. Under these conditions, the overall detection efficiency for a full cascade ε_c becomes proportional to the total cascade energy E_c and independent of the specific cascade path:

$$\varepsilon_c(E) = 1 - \prod_i (1 - \varepsilon_i(E)) \approx \sum_i \varepsilon_i(E) = k \sum_i E_i = k \cdot E_c \qquad (3)$$

The efficiency of C_6D_6 detectors is not inherently linear and the counts must be weighted by an energy-dependent factor, called weighting function (WF). By defining the detector response function $R(E_d, E_\gamma)$ the fraction of γ-rays of energy E_γ that leave an energy deposit ("pulse height") E_d in the detector, and i discrete form R_{ij}, the weighting function $W(E)$ (W_i in discrete form) is applied to each detected event based on the deposited energy. The response functions R_{ij} are obtained using Monte Carlo simulations with codes like MCNP and Geant4.

Status of ^{63,65}Cu measurements

In 2024, the neutron capture measurements on ^{63}Cu and ^{65}Cu were carried out using the setup shown in Figure 2. It consisted of four C_6D_6 liquid scintillation detectors, arranged around the copper sample, and a neutron beam monitor was positioned at the entrance of the beamline. The analysis of the copper data is underway. Following the energy calibration, the focus is now on the construction of the weighting functions, a crucial step in using the C_6D_6 setup.

Capture data requires also the measurements of the total, elastic, and inelastic cross sections. The total cross section is determined from transmission measurements, by comparing the TOF spectra recorded with and without the sample in the beam path. In the keV to MeV energy range, flux measurements can be performed using a fast ionization chamber loaded with ^{235}U.

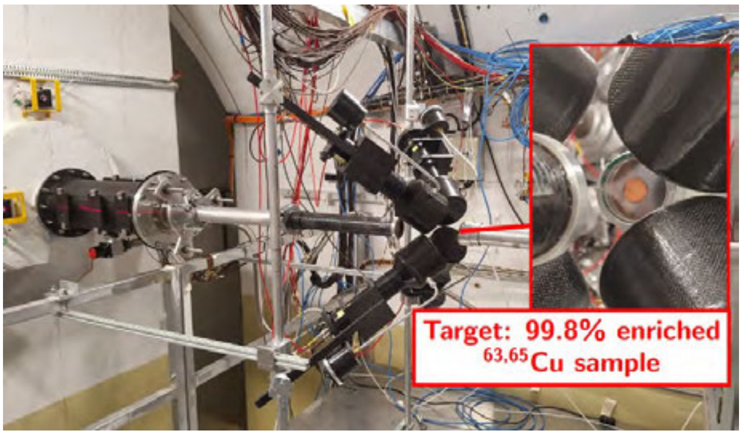

Fig.2: Image of the setup used in 2024 for the capture measurement campaign, with the four C6D6 detectors surrounding the Cu sample target.

For inelastic and elastic channels, a new detection setup has been developed and tested at n_TOF. It is based on an array of eight stilbene scintillators which demonstrated excellent performance in neutron-gamma discrimination time and resolution, making them suitable for differential cross section measurements of both elastic and inelastic processes [4].

Conclusions

The n_TOF at CERN provides high-precision measurements of neutron-induced reaction cross sections over a wide energy range. Successful tests demonstrated the capability of n_TOF to measure total, elastic, and inelastic cross sections. This enables a comprehensive characterization of $^{63,65}Cu$ isotopes, relevant to both advanced reactor technologies and nuclear astrophysics. Improved copper cross section data are crucial for enhancing the simulation accuracy of experiments conducted at the TAPIRO research reactor at ENEA, an important facility for the development and validation of Generation IV fast reactor systems, such as Lead-cooled Fast Reactors (LFRs). These new measurements will directly contribute to reducing uncertainties in reactor modelling and nuclear data libraries, supporting the advancement of safe and efficient reactor technologies.

Acknowledgment

Authors acknowledge the Italian National Institute for Nuclear Physics (INFN), Prof. Cristian Massimi (Dept. of Physics and Astronomy, University of Bologna) and the n_TOF Collaboration at CERN.

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ROUNDTABLES

SEVERE ACCIDENT ANALYSIS FOR ADVANCED REACTORS: SAFETY-BY-DESIGN, MITIGATION STRATEGIES, AND REGULATORY CHALLENGES

Chair: Fulvio Mascari

Participants:

Prof. Sandra Dulla	University of Pisa
Prof. Andrea Pucciarelli	University of Pisa
Prof. Fabio Giannetti	University of Rome
Dr. Guido Gerra	*new*cleo

The session titled *"Severe Accident Analysis for Advanced Reactors"* explored safety-by-design principles and the consequent role of severe accident analysis in assessing mitigation strategies and addressing associated regulatory challenges in the deployment of advanced reactor designs, with a particular focus on Small Modular Reactors (SMRs) and Advanced Modular Reactors (AMRs). The key ideas discussed during the discussion panel are reported below.

Severe accident analyses play a key role in demonstrating that, even in extreme, low-probability events characterized by multiple failures, plant design provisions are available to ensure the protection of workers, the public and the environment. This is equally important for advanced reactors, where new design solutions, such as passive safety systems, may replace traditional ones, such as active safety features. In fact, even when the first three levels of Defence-in-Depth (DiD) are reinforced, for example through the adoption of passive safety systems, a robust demonstration of the reactor's capability to cope with postulated severe accidents is still required, addressing DiD levels 4 and 5.

Light Water SMRs are derived from well-proven Light Water Reactor (LWR) technologies, benefiting from decades of operational experience and feedback. They include evolutionary features (e.g. passive safety systems) aimed at enhancing the plant's inherent safety. In such cases, the definition of a severe accident remains similar to that applied to current operating reactors. Existing analysis methods and tools can still be applied; however, they must be updated and supported by specific experimental programs and may be integrated with advanced methodologies to assess the new features and provide validation aimed at characterizing their evolutionary aspects. AMRs, on the other hand, often introduce innovative features (such as different working fluids, materials, fuel types etc.) that depart from current operating reactors. These innovations may require new definitions of what constitutes a severe accident, as well as the identification of credible scenarios that could lead to such conditions. Owing to changes in, e.g., working fluids and geometries compared to current operating reactors, close collaboration with licensing authorities will be necessary to assess whether existing modelling techniques are suitable, whether they need to be extended, validated and verified for new applications, or whether entirely new approaches and methodologies will be required to achieve accurate predictive capabilities.

In this context, whenever innovative solutions are introduced or established engineering practices are exceeded, their safety performance and reliability must be demonstrated through a combination of targeted research, performance testing with clearly defined acceptance criteria, and operational feedback from other relevant applications. These features should be tested before deployment and then monitored during operation to ensure that their behaviour aligns with safety expectations. Another key point is the demonstration of "practical elimination" of accident sequences that could lead to early or large radiological releases. In many regulatory environments (and according to IAEA guidelines), practical elimination must be sought if and only if the given accident scenario cannot be practically mitigated. This becomes a crucial requirement for AMRs, in which the limited magnitude of the source term (compared to traditional power reactors) often implies a greater ability to mitigate the consequences of a severe accident (and therefore removes the need/licensing basis for practical elimination).This cannot be claimed without being supported by a solid and rigorous technical safety justification, especially when innovative systems are involved. For both SMRs and AMRs, introducing innovations requires more than just modelling. Experimental testing is therefore essential to confirm system behaviour and performance, reduce uncertainties, and validate computational tools. With regard to these tools, the primary means of validation is the experimental evidence. Consequently, a combined approach involving both experimental data and analytical methods is confirmed based on current practice to be necessary to develop a robust safety demonstration.

As a final consideration, particular attention must be given to common cause failures, including those resulting from external hazards. Since multiple modules often share safety-related systems, the simultaneous failure of several units could lead to consequences that exceed the assumptions made for a single module. Therefore, robustness against common cause failures and the related risks must be clearly assessed and appropriately addressed, especially if the severe accident analyses and considerations are based only on a single module.

EXPERIMENTAL DATA FOR INNOVATIVE NUCLEAR REACTORS

Chair: Marco Utili

Participants:

Dr. Mariano Tarantino	ENEA
Dr. Fabio Moretti	*new*cleo
Prof. Giuseppe Nallo	*new*cleo/Politechnic of Turin
Prof. Stefano Lorenzi	Politechnic of Milan

M. Utili introduces one of the most complex tasks for the development of the innovative fission reactors (SMR or GEN-IV reactors), the availability of a Nuclear Experimental Database. we are moving in a sector where all the new components and systems developed are prototypes and require experimental validation. The components and systems developed for the Innovative nuclear reactor, particularly LFR, will operate in lead under relevant neutron flux, where only a few data are available. Therefore, the experimental qualification is mandatory to consider the various factors that can play a relevant role in the operation of the reactor. The experimental validation must be carried out at all levels: materials, components (taking into account additional complexity as manufacturing procedures, like welding or connection), systems validation (thermohydraulic, structural integrity, and main function of system or components), and procedure validation in the relevant environment.

Several codes are available for the Thermo-fluid dynamics, Thermo-mechanic, neutronic, and safety analysis, but combining all these phenomena is quite complex, increasing the complexity of the project. Moreover, in the case of AMR, it is necessary to include also the chemistry of lead.

Only a few facilities in the world are available for the experimental validation of samples and components in lead under a relevant neutron flux.

Multiphysics codes that take into account all the phenomenology and the interaction among them are under development, but in some case, no experimental data are available for the implementation; therefore, a strong interaction among the designer, numerical code developer, and experimentalist is requested to identify the missing data and how to design the experiments with the required accuracy.

Discussion:

Dr. M. Tarantino provides an overview of the main international experimental infrastructures available for the Lead Fast Reactors components and system validation. An analysis of the roadmap in Italy and Europe for the experimental data production and management is carried out. The main challenges for the adaptability of experimental databases developed for Generation III reactors to SMR and AMR reactors were discussed. The analysis considers how to address the issue of time and budget constraints.

Dr. Fabio Moretti showed the fast-track R&D programme of *new*cleo to support the development and the safety demonstration of LFRs. The programme is already bringing tangible technological advancement, know-how, and precious data. Much more to come within the next few years. This is made possible notably by a huge effort, resources, motivation from *new*cleo and strategic collaboration with ENEA (and, particularly, the Brasimone Research Centre). The main facilities developed in collaboration with ENEA were discussed within the methodologies applied and the main difficulties in the design of an experiment for the thermo-fluid dynamic qualification of components or systems for LFR.

Prof. Giuseppe Nallo shows the strategies implemented by *new*cleo for the qualification of scientific computing tools, which strongly leverage the experimental R&D programme. The tool starts with the Input analysis, Phenomena Identification and Ranking Table Process (PIRT) analysis, Validation and Verification matrix definition within the analysis of the main gaps, performance assessment and transposition with the final validation. This approach is applied to all components and systems of the reactor. PRECURSOR, a full-height ITF experimental facility that will be available in ENEA Brasimone in 2026, will be used for the code validation. The facility is characterised by a small power-to-volume scaling factor (1:9) representative of the LFR-AS-30 in terms of 3D TH behaviour, with emphasis on natural circulation. PRECURSOR features electrically heated fuel elements and comprises both the primary side and the secondary side, including the balance of plant (turbine, condenser etc..).

The strategy and main difficulties in the validation of thermos fluid dynamic codes for the innovative Lead Fast Reactor were discussed.

Prof. Stefano Lorenzi introduced the limits of applicability of Multiphysics codes for LFR reactor core design concerning experimental validation.

POSTER SESSION

EXPERIMENTAL ACTIVITIES USING SOLEAD FACILITY

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Abstract

SOLEAD (Solar tOwer LEAd Demo) is an experimental facility initially designed to carry out experiments relevant to Concentrated Solar Power (CSP) air-based tower systems. However, due to its versatility, the facility has been used to carry out tests of interest for fast reactors LFR type. The facility is mainly composed of a storage tank, a main vessel and a gas panel. The storage tank was used for corrosion tests, in static conditions, on structural material specimens. Tests, consisting of steam injection inside the liquid lead, have been performed to support the safety analysis (leak-before-break in a steam generator tube). Tests have been performed to support the safety analysis concerning the steam inlet in cover gas during the fuel assembly handling. Air injection tests in the cover gas had the scope to study the kinetics of the oxygen diffusion from the cover gas to the liquid lead. Tests of interactions between air and liquid lead are currently in progress, to study the oxygen diffusion inside the lead. Tests for the characterisation of different types of oxygen getters are also foreseen. All these tests require an accurate knowledge of the oxygen concentration inside the lead, which is measured using one or more electrochemical oxygen sensors. Also the main vessel has been used to test a prototypical pin of the core simulator installed in ATHENA experimental facility. This work contains a brief description of the experimental facility, of the experimental campaigns concluded, in progress and planned.

Keywords: *SOLEAD; ATHENA; LFR; Oxygen Control.*

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A WELL-ESTABLISHED METHODOLOGY FOR THE CONCEPTUAL DESIGN AND NEUTRONIC ANALYSIS OF ACCELERATOR-DRIVEN SYSTEMS (ADS), BASED ON BOTH DETERMINISTIC AND MONTE CARLO CODES

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Abstract

Aiming at the closure of the fuel cycle to enhance the sustainability of nuclear energy, over the past thirty years the ENEA Laboratory for the Design and Analysis of Nuclear Systems has been engaged in studies related to the Accelerator Driven System (ADS). The initial impetus for feasibility studies of these innovative concepts was provided by C. Rubbia (President of ENEA in 1999-2005), followed by numerous research activities within Euratom projects, national programmes and international collaborations. As notable examples of lead-cooled ADS conceptual designs, based on an 800 MeV proton accelerator sustaining a ~400 MW_{th} sub-critical core:

- in the EU EUROTRANS project, a minor actinides (MA) burner loaded with a U-free binary fuel - TRU (Pu and MA) - within a MgO inert matrix was designed to operate in an open fuel cycle;
- in a collaborative project promoted by IAEA on the LEU ADS, a MA burner loaded with a ternary fuel (Th, U, TRU) was defined to operate in a Th-U closed fuel cycle.

More recently, within a project agreement between ENEA and Transmutex (a Swiss company), an ADS loaded with metallic Th (and TRU) fuel - operating in a closed fuel cycle, with the dual purpose of burning TRU (from UOX spent fuel) and ^{233}U breeding - has been preliminarily investigated. The well-established ENEA methodology - developed for the sub-critical core design and neutronic analysis, relying on both deterministic (as ERANOS) and Monte Carlo (as MCNP and GEANT4) codes - was adopted, and here briefly discussed.

Keywords: *Accelerator Driven System (ADS); Sub-critical core Design; Neutronic analyses; Deterministic and Monte Carlo methods; Minor Actinides/TRU burning.*

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MONTE CARLO METHOD AS A NUMERICAL TOOL TO SUPPORT THE DESIGN OF NUCLEAR POWER REACTORS AND RESEARCH FACILITIES

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Abstract

The Monte Carlo method is a reference for nuclear particle transport simulations, since the inherent stochastic nature of particles behaviour is effectively modelled through a probabilistic approach. Thus, Monte Carlo codes are fundamental tools for nuclear systems analysis and experiments preparation. Results of interest are mainly response functions, evaluated for certain particle species at some volumes in the phase space of the system. Simulations may regard either fixed source problems or eigenvalue problems. The first kind presents a given source of particles: the transport equation has a source term. On the other hand, the eigenvalue problems concern systems made of multiplicative media – i.e., including fissile materials – in which the source depends on the system itself and consists of the first eigenfunction of the homogeneous transport equation. Thus, the source is preliminary defined, the eigenfunction obtained and the related eigenvalue is referred to as multiplication coefficient of the system; the response functions of interest are evaluated afterwards. Eigenvalue problems have been solved for nuclear power reactor design - such as Gen-III (EPR), Gen-IV LFRs and research reactors (TAPIRO, ZED-II). Flux distributions have been obtained, and sensitivity and uncertainty analysis considering the multiplication coefficient performed with respect to some parameters of interest (e.g., cross sections). Considering fixed source problems, analysis of irradiation experiments for n_TOF facility at CERN and studies regarding subcritical systems (ADS) are presented. Monte Carlo simulations are performed through useful codes for nuclear physics and technology such as MCNP6.3, OpenMC and Geant4.

Keywords: *Monte Carlo; multiplication coefficient; criticality; stochastic methods; nuclear system simulations*

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EXPERIMENTAL FACILITY *NEW*CLEO BRASIMONE FOR CORROSION TESTS IN LIQUID LEAD

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Abstract

Qualification and standardization of structural and core materials are fundamental aspects of the *new*cleo project. Given the highly aggressive environment of Lead-cooled Fast Reactors (LFRs), material selection must meet multiple criteria - most notably, resistance to heavy metal corrosion and liquid metal embrittlement. Demonstrating these properties is a prerequisite for submitting material data to regulatory authorities as part of the reactor prototype licensing process. To support these efforts, *new*cleo has developed facilities in Brasimone site to characterize material properties in liquid Pb. For this reason, the first two facilities, are focused on addressing the corrosion behavior of various materials.

- **CAPSULE**: A standardized stagnant lead cell equipped with a bubbling gas injection system for oxygen control. This facility enables sensitivity analyses of corrosion and oxidation rates on various samples by adjusting parameters such as temperature, exposure duration, and oxygen content.
- **CORE**: A loop-type facility where lead is circulated by a pump. It includes dedicated sections for corrosion testing, erosion testing, and filtration. An integrated oxygen control system ensures consistency across tests.

To address additional mechanical properties, other setups focused on investigating liquid metal embrittlement through creep, fatigue, tensile tests are currently being commissioned. To reduce data variability, standardizing experimental setups, testing protocols, and analysis methods is key to ensuring reliable, comparable, and regulatorily accepted results. Highlights of these activities will be presented in this presentation.

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TOWARD HIGH-PERFORMANCE STRUCTURAL MATERIALS FOR THE NEXT GENERATION OF NUCLEAR REACTORS

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Abstract

Next-generation reactors promise enhanced sustainability, safety, and fuel efficiency; however, their deployment is significantly constrained by the performance of structural materials under extreme operating conditions. Key limitations—such as high-temperature mechanical strength, radiation resistance, compatibility with aggressive coolants (e.g., lead or molten salts), and long-term microstructural stability—pose major challenges to the qualification and implementation of materials in advanced nuclear systems, including Generation IV lead-cooled and molten salt fast reactors (LFR, MSR), as well as the fusion reactor technology. In this context, candidate materials such as FeCrAl ODS alloys, AFA steels, and advanced protective coatings (e.g., alumina or Al-based surface alloying) have shown promising potential in enhancing corrosion resistance while preserving mechanical integrity at elevated temperatures. Advancing these materials toward industrial application requires an increase in their Technology Readiness Level (TRL), which requires rigorous multi-scale characterization and alignment with demanding industrial codes such as RCC-MRx and ASME BPVC.

Since accelerating the qualification process is essential to foster timely innovation, European research efforts are converging on integrated strategies that support regulatory compliance, the codification of advanced structural materials, and foster synergies between fission and fusion applications. Within this framework, ENEA is involved in European projects such as INNUMAT (Innovative Structural Materials for Fission and Fusion, running from September 1, 2022, to August 31, 2026), and CONNECT-NM, a new European co-funded partnership aiming to accelerate research and innovation by uniting a wide network of stakeholders to address key areas such as material development, qualification, standardization, non-destructive evaluation, and digital knowledge management.

Keywords: *Gen-IV reactors, advanced materials, TRL, material qualification, material characterization.*

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DEMAGAGGER, A TOOL FOR OPTIMIZING THE COOLANT FLOW DISTRIBUTION IN CORE CONFIGURATIONS BASED ON CLOSED ASSEMBLIES

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Abstract

The thermal hydraulic design of a nuclear fast reactor core has, as one of its main objectives, the respect of limits on the maximum temperature of critical components. The typical strategy to tackle this is to optimize the power-to-flow ratio of assemblies, distributing them in cooling groups. DEMAGAGGER has been developed by the NUC-ENER-PRO laboratory as a tool designed to compute an optimized cooling group scheme for a fast nuclear reactor. The tool requires, as input, information concerning core-wise power distributions (usually computed by neutronic codes), geometrical properties of the core, thermal hydraulic data (inlet coolant temperature and global mass flow rate) and hot spot factors. The latter are used to compute a cooling group scheme that can guarantee an adequate cooling of the core in normal conditions with quantified uncertainties.

An example of application of DEMAGAGGER on the ALFRED core is here presented. Several core-wise power distributions, representative of begin and end of cycle of a multi-batch fuel scheme, computed using the ERANOS code, are used as input data, together with hot spot factors computed by the AMACA tool. Results are provided showing the effect of several cooling group schemes on core-wise sub-channel bulk temperature distributions. However, the combined effect of power distributions and hot spot factors leads to a non-feasible cooling group definition since the required minimum mass flow rate is greater than the one available. To overcome this issue, improvements in the core design must be considered, but also a study for decreasing the uncertainty sources.

Keywords: *Core thermal hydraulics design; Cooling groups; Hot channel and hot spot factors*

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EMISSIVITY OF MOX FUEL: A REVIEW OF EXPERIMENTAL RESULTS

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Abstract

Thermal radiation properties of nuclear fuel, especially at high temperatures, are relevant to safety analysis. For this purpose, the spectral emissivity of fuel should be determined well beyond the melting temperature. Emissivity is defined as the ratio of the emissive power of a body to the emissive power of a blackbody at the same temperature. The total hemispherical emissivity is relevant to determining the thermal radiation from a molten core. The spectral emissivity is also necessary for reliable pyrometric measurements of fuel temperature, especially close to phase transition. A review of experimental results published in the open literature has been performed. Recent results showed to be in agreement with recommendations provided by Bober and his co-authors.

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LFR SAFETY ACTIVITIES: CIRCE-THETIS PRE-TEST AND TRANSIENTS ANALYSIS

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Abstract

ENEA is one of the main actors in the Lead Fast Reactor (LFR) technology development: the recently started Horizon Europe project LESTO is coordinated by ENEA and confirms its leading role in the LFR community; the state-of-the-art experimental facilities in Brasimone Research Center provide valuable data in the fields of thermohydraulic, innovative materials, coolant chemistry control, etc. Among LFR related activities, ENEA is performing pre-test calculations for the CIRCE-THESIS experimental campaign, which aims at investigating thermal-hydraulic behavior, in both steady state and transient conditions, including natural circulation, and characterizing the performance of a prototypical helical coil steam generator. SIMMER is widely considered the reference code for transient analysis of liquid metal cooled systems. Its flexibility allows simulating a wide range of accidental scenarios with multi phase and multi component capabilities. For these unique features, SIMMER is regarded as the primary option for the safety assessment of LFR systems. Therefore, the development of more detailed and refined models to simulate the specificities of the LFR systems is essential, and so is its validation, as already foreseen in LESTO. After establishing a steady state 2D model of the facility, including its fuel pin simulator, heat exchanger and layout, we simulated both the loss of flow and the loss of heat sink, both protected and unprotected. The outcomes are mostly in line with expectations, proving the suitability of the SIMMER code and the models for LFR applications, and call for further investigations when the experimental data will be available.

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FINAL REMARKS AND PERSPECTIVES

Through a series of technical presentations, interactive discussions, and dedicated roundtables, the SERAFIN Workshop represented an important opportunity to consolidate existing collaborations and foster new synergies among the research groups involved in nuclear science and technology. The event enabled a comprehensive exchange of experiences, methodologies, and perspectives on the development of innovative nuclear technologies, contributing to a shared and cohesive vision to address current and future challenges in the field.

The workshop was attended by a broad and diverse audience, comprising more than eighty participants from research institutions, universities, and industry. In particular, the event gathered researchers from ENEA, representatives from industrial partners such as *new*cleo, as well as academics, PhD candidates, and students from several Italian and international universities. In addition to on-site participation, a number of attendees joined the workshop remotely, further extending the reach of the discussions and facilitating broader engagement. The distribution of contributions across the four thematic sessions reflected the multidisciplinary nature of the workshop and highlighted the strong interaction between experimental research, modelling activities, technological development, and applications.

A special acknowledgment is due to the University of Bologna for hosting the workshop and for its continuous support in the organization of the event. The long-standing and fruitful collaboration between ENEA and the University of Bologna was once again confirmed as a cornerstone for research, education, and innovation in the nuclear field, fostering the integration of academic excellence with applied research and technological development.

A particularly valuable contribution to the workshop was provided by the two roundtables, which allowed an open and in-depth discussion on key cross-cutting topics of strategic relevance. The debates on severe accident analysis for advanced reactors and on the role of experimental data for innovative nuclear systems stimulated constructive exchanges among experts from different backgrounds, leading to meaningful reflections and highlighting priority areas for future research and collaboration. These moments of discussion complemented the technical sessions and reinforced the role of the workshop as a forum for strategic thinking beyond individual research activities.

The success of the SERAFIN Workshop was made possible by the active participation, commitment, and constructive spirit of all attendees. The feedback collected during the event confirmed the relevance of the workshop format and its potential to further evolve in the coming years. Building on the outcomes of this edition, future initiatives will continue to strengthen collaboration among research institutions, universities, and industry, supporting the advancement of nuclear technologies in line with national and international objectives for safety, sustainability, and innovation.

Highlights from the Event


ENEA's core design approach
The enabling tools
ENEA


Simulated cases
ENEA


2) Elaborazione di proposte:
Lista delle priorità da affrontare per lo sviluppo di ciascuna filiera selezionata:
Key issue per attività di sviluppo/qualificazione sulle filiere
Ricognizione delle competenze e capacità italiane
Collaborazioni internazionali che possono fornire sinergie
3) Roadmap e linee guida:
Per le tecnologie selezionate:
Tempi e modalità di attivazione dei prototipi e dei primi reattori di produzione;
Identificazione e tempistica di soluzione per le criticità tecniche evidenziate;
Stima dei costi di R&S e qualificazione delle soluzioni selezionate;
Diagramma temporale con principali tappe per l'attivazione delle filiere in esame;
Identificazione e ipotesi di soluzione delle criticità generali e sistemiche nel settore nucleare (autorità di sicurezza, formazione, cornice giuridica e legale ecc..) in interazione con l' attività dei GdL 4 e 6.
MASE
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RSE


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Enabling advanced thermal-hydraulics knowledge: the NETT1 Laboratory


1950s–60s The Origins
• USA: EBR-I (1951), first sodium-cooled reactor to generate electricity
• Global interest: USSR, UK, France, Germany, Japan
France
• Rapsodie (1967): experimental prototype
• Phénix (1973): demonstrator, closed fuel cycle
• Superphénix (1985–1997): 1200 MWe
Italy
• PEC (Brasimone): fuel testing for SPX, never completed
Russia
• BN series: BN-350, BN-600 (still operating), BN-800
1990s Decline Shutdowns due to cost, accidents, and public opposition
2000s Generation IV Revival SFR selected among the 6 GIF advanced reactor concepts
• Ongoing projects: ASTRID (France, canceled), PFBR (India)
• European projects: CP-ESFR, ESFR-SMART (full size), ESFR-SIMPLE

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