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Joint European Canadian Chinese development of Small Modular Reactor Technology

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Executive Summary

The workshop on “Structural Materials for Gen IV SMR Concepts” held in November 2023 in Madrid, Spain, was one of the workshops in the ECC-SMART project workshop series and considered one of the key milestones for the ECC-SMART project.

The workshop hosted by the ECC-SMART project in collaboration with the EERA-Joint Programme of Nuclear Materials and the Generation IV International Forum (GIF) aims to foster an exchange of knowledge and ideas among scientists and engineers specializing in fourth-generation reactor designs. This event focused on the advanced materials needed to endure extreme conditions such as high temperatures, pressure, radiation and corrosion which are characteristic of Generation IV reactors.

The Workshop was designed not only for seasoned professionals, but it also welcomed young researchers and engineers eager to deepen their understanding and engage with the latest innovations in the field. This workshop served as a vital forum for networking, collaborative learning and exposure to innovative research. Attendees benefited from discussions on the latest developments, tackled current challenges and explored innovative solutions in reactor materials.

Participation in this workshop offered a unique opportunity to the experts and young generation to connect with each other, gain fresh insights and contribute to the progress of material technologies essential for the next generation of nuclear reactors.

The workshop took place under the implementation of Task 6. Work Package 6 of the ECC-SMART project. This event served as a platform for sharing the findings of the ECC-SMART project with a broader audience. Contributors emphasized the commitment of the project to transparent communication of its scientific advancements. It can be concluded that the workshop successfully fulfilled the dedication of the project to education and training.

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List of acronyms and abbreviations:

Acronym/Abbreviation	Definition
AFAs	Alumina-forming Alloys
AFM	Atomic Force Microscopy
AM	Additive manufacturing
AMME-WG	Advanced Manufacturing and Materials Engineering Working Group
CHT	Conjugate Heat Transfer
CIEMAT	Spanish National Research Centre for Energy, Environment and Technology
CNL	Canadian Nuclear Laboratories
CNTs	Carbon nanotubes
CO	Confidential
CVR	Centrum Výzkumu Řež s.r.o.
CYP	Cyclic Potentiodynamic Polarization
DNS	Direct Numerical Simulations
EDS	Energy Dispersive Spectrometry
EDX	Energy Dispersive X-ray Analysis
ECC-SMART	The Joint European Canadian Chinese development of Small Modular Reactor Technology
EELS	Electron Energy Loss Spectroscopy
EIS	Electrochemical Impedance Spectroscopy
ENEN	European Nuclear Education Network
EU	European Union
EU-CON	Classified Information: CONFIDENTIEL UE
EU-RES	Classified Information: RESTREINT UE
EU-SEC	Classified Information: SECRET UE
H2020	Horizon 2020 (EU Framework Programme for Research and Innovation)
HLM	Heavy Liquid Metal
HTGR	High Temperature Gas-cooled Reactor
JRC	Joint Research Centre
KIT	Karlsruhe Institute of Technology
LME	Liquid Metal Embrittlement
LPR	Linear Polarization Resistance
MQL	Minimum Quantity of Lubricant
MSRs	Molten Salt Reactors
NEWCLEO	NEWCLEO Corporation
OCP	Open Circuit Potential

OECD/NEA	Organisation for Economic Co-operation and Development / Nuclear Energy Agency
PEO	Plasma Electrolytic Oxidation
PM HIP	Powder Metallurgy Hot Isostatic Pressing
PNNL	Pacific Northwest National Laboratory
PU	Public
SC	Supercritical
SC-CO2	Supercritical Carbon Dioxide
SCWR	Supercritical Water-cooled Reactor
SEM	Scanning Electron Microscopy
SLM	Selective Laser Melting
SMR	Small Modular Reactor
SPM	Scanning Probe Microscopy
SSRT	Slow Strain Rate Tests
TEM	Transmission Electron Microscopy
VHTR	Very High Temperature Reactor
WDS	Wavelength Dispersive Spectrometry

Annex A

List of appendices

1 Introduction

1.1 Project summary

The Joint European Canadian Chinese development of Small Modular Reactor Technology (ECC-SMART) is a multinational project focused on the development and licensing of the future Supercritical Water-cooled Small Modular Reactor (SCW-SMR).

The ECC-SMART project main objectives are:

- to provide science-based recommendations and methodologies for performing safety evaluations and safety improvements fostering the safety standards, including the experimental validation of essential items for safety demonstrations related to the SCW-SMR,
- to derive the most economical safety-driven SCW-SMR design requirements by identifying adequate solutions to key technical issues which drive cost and safety and their influence on the future licensing process,
- to increase the level of knowledge as well as the interest of the industrial partners and demonstrate the benefits of the SCW-SMR concept.

1.1.1 Purpose of this document

This document is the second deliverable of the Work Package 6 “Dissemination and communication” lead by ENEN and the second in series of the proceedings to be issued following the project workshops: (1) Workshop on thermal hydraulics and neutronics of the SCW-SMR, (2) Proceedings of workshop on materials and online in-situ methods for corrosion monitoring in SCWR and (3) Proceedings of the ISSCWR. This document thus provides the Proceedings of the workshop held in Madrid, Spain, in November 2023. The deliverable aims at the following objectives:

- to share insights, methodologies and best practices.
- to promote, sustain and increase the stakeholders’ awareness on the on-going efforts to study the structural materials for Gen IV SMR Concepts.

2 Workshop Programme

The Workshop on the Structural Materials for Gen IV SMR Concepts, held in Madrid, Spain, in November 2023, focused on the structural materials and online in-situ methods for corrosion monitoring in SCWR and including the latest findings from the other advanced Gen IV. concepts (Ref.: Fig.1). The three-day event featured a blend of in-person and virtual participation. The workshop began with an opening session, setting the purpose and welcome words from the organizing team.

Involved organizations CIEMAT, OECD/NEA, PNNL, CVR, CNL, NEWCLEO, TEXAS UNI, ENEN covered activities related to materials for Generation IV SMRs and structural materials for nuclear applications, provided insights on the Generation IV International Forum (GIF), emphasized the international collaborative efforts, presented ECC-SMART project initiatives and the outlines of its main research activities, shared information and opportunities of the new research programs, provided Euratom financial mechanisms of mobility opportunities.

Daily topics covered the following:

Day 1 (27 Nov 2023): Session 1 focused on introductions and research activities in CIEMAT. Session 2 covered structural and cladding materials for Gen IV reactors, including design aspects of SCWR and materials compatibility with heavy liquid metal coolant.

Day 2 (28 Nov 2023): Session 3 explored the influence of environment, coolant and radiation on structural and cladding materials for SCW-SMR. Session 4 discussed the advanced microstructure examination and testing methods to support Gen IV SMRs materials research. Session 5 covered the development of advanced materials solutions, fabrication processing, and other related topics.

Day 3 (29 Nov 2023): Session 5 continued discourse on the development of new grade heat-resistant iron-based alloys, advanced machining processes and data management services in support of the European Engineering Materials Community.

The workshop provided a comprehensive platform for exchanging knowledge on the latest developments in materials field for Gen IV nuclear reactors and, highlighted significant international collaborations and innovations in materials science applicable to nuclear energy.



Fig. 1: Workshop participants; Source: ECC-SMART Project.

The Programme of the workshop is provided in Annex A, appendix 1.

Below is the detailed description of the contributions of the Workshop, outlining the key topics addressed, the insights shared by various experts and a summary of the significant discussions and outcomes that emerged from the sessions.

All presentations are provided in Annex A, appendix 2.

2.1 SESSION 1: INTRODUCTION

2.1.1 Introduction to GIF and the OECD NEA activities related to SMR EU, Elena Poplavskaia, OECD NEA

The presentation "Introduction to GIF and the OECD NEA activities related to SMR EU" by Elena Poplavskaia provides an overview of the Generation IV International Forum (GIF) and its activities related to the development and deployment of advanced nuclear reactor technologies. It covers the structure of GIF, including its various working groups and their focus areas such as safety, proliferation resistance, economic modeling, and advanced manufacturing. The presentation also discusses the challenges of transitioning from research and development to demonstration and deployment of these technologies to meet net-zero goals.

2.1.2 Overview of the GIF-Advanced Manufacturing and Materials Engineering Working Group (AMME-WG), Isabella van Rooyen, Co-Chair AMME-WG

The presentation "Overview of the GIF-Advanced Manufacturing and Materials Engineering Working Group (AMME-WG)" by Isabella van Rooyen discusses the activities of the GIF Advanced Manufacturing and Materials Engineering Working Group (AMME-WG). It covers the transition of AMME from a Task Force to a Working Group, its objectives, focus on advanced materials and manufacturing technologies and planned activities aimed at reducing deployment time for Generation IV reactors. The presentation also outlines the importance of industry collaboration in accelerating the qualification of new manufacturing processes and materials.

2.1.3 ECC-SMART Project, Michaela Krydova, CVR

The presentation "ECC-SMART Project" by Michaela Krydova focuses on advancements in structural materials for Generation IV small modular reactors (SMRs) as part of the ECC-SMART project. It details the project funding, goals and the consortium involved, highlighting efforts in materials testing, thermal hydraulics, neutron physics and safety standards. The presentation also reviews recent progress and publications related to these areas.

2.1.4 Coordination of the European Research Community on Nuclear Materials for Energy Innovation – CONNECT-NM

The presentation "Coordination of the European Research Community on Nuclear Materials for Energy Innovation – CONNECT-NM" by Lorenzo Malerba discusses the proposal for a co-funded European partnership under the Euratom work programme 2023-2025. It emphasizes innovative approaches to develop and qualify new nuclear materials and advanced manufacturing processes, aiming to shift from traditional methodologies to more integrated and predictive models for material behavior in nuclear environments.

2.2 SESSION 2: STRUCTURAL AND CLADDING MATERIALS FOR GENIV REACTORS (INCLUDING SMRs)

2.2.1 "The Super Critical Water-Cooled Nuclear Reactor", Armando Nava, CNL

The presentation "The Super Critical Water-Cooled Nuclear Reactor" by Armando Nava discusses the development and goals of Gen-IV supercritical water-cooled reactors (SCWRs), highlighting the Canadian SCWR. It explores the advantages of SCWR technology, including increased thermal efficiency and the integration of lessons from over 70 years of water-cooled reactor operations and supercritical fossil-fired plants. The presentation outlines the design features, reactor core components, R&D initiatives, opportunities and challenges associated with SCWRs.

2.2.2 Integrated Material and Advanced Manufacturing Development of Structural Materials: US DOE AMMT Program, Isabella van Rooyen, Co-Chair AMME-WG

The presentation "Integrated Material and Advanced Manufacturing Development of Structural Materials: US DOE AMMT Program" by Isabella J. van Rooyen delves into the Advanced Materials and Manufacturing Technologies (AMMT) Program led by the US Department of Energy. It outlines the strategies for developing and deploying structural materials through integrated material and manufacturing approaches, focusing on the Generation IV reactor needs. It covers the methods and progress of the US Department of Energy efforts in enhancing structural materials for nuclear applications. The focus is on how these developments under the Advanced Materials and Manufacturing Technologies (AMMT) Program can affect various types of reactors. It discusses the critical advancements in materials science, including the design and testing of materials that can endure extreme conditions which is crucial for the next generation of nuclear reactors. The presentation also highlights partnerships with national laboratories and industry stakeholders aiming to streamline the path from material innovation to practical application to ensure reactors are safer and more efficient.

2.2.3 Qualification of materials for compatibility with HLM coolant, Serguei Gavrillov, SCK CEN

The presentation "Qualification of Materials for Compatibility with HLM Coolant" by Serguei Gavrillov (Ref.: Fig. 2) discusses the evaluation and selection of structural materials for their compatibility with heavy liquid metal (HLM) coolants focusing on their use in Generation IV Small Modular Reactor (SMR) concepts. This research is crucial for advancing nuclear reactor designs that utilize HLM coolants like lead-bismuth eutectic (LBE) which pose specific challenges such as liquid metal embrittlement (LME) and corrosion. It reviews experimental methodologies like slow strain rate tests (SSRT) and corrosion assessments to determine the material properties after exposure to HLM. The objective is to establish robust materials that can withstand the specific challenges posed by HLM to enhance the safety and efficacy of Generation IV nuclear reactors. The presentation details ongoing efforts to understand and mitigate these effects to ensure the materials integrity and performance in nuclear environments.

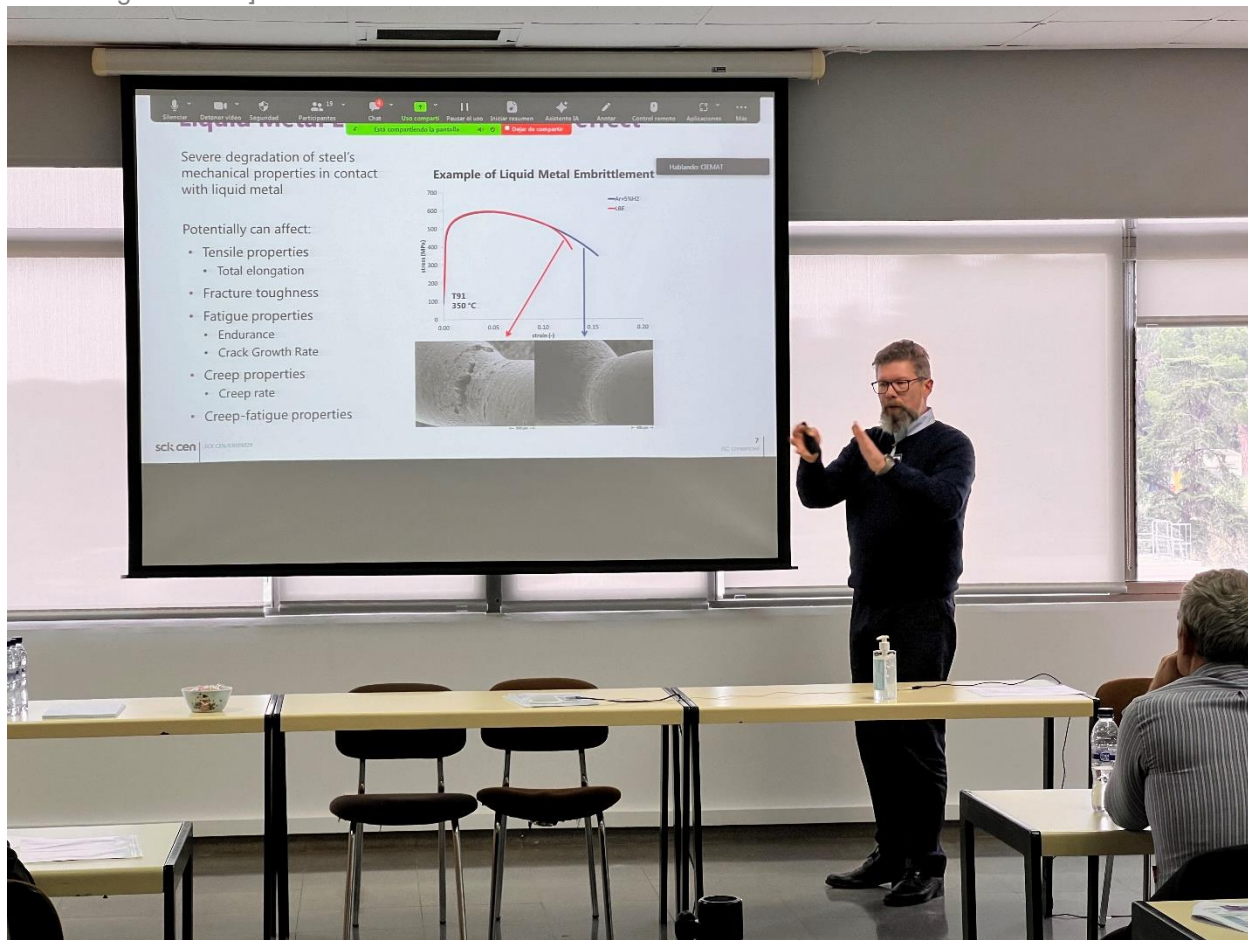


Fig.2: Qualification of materials for compatibility with HLM coolant, Serguei Gavrillov, SCK CEN; Source: ECC-SMART Project

2.2.4 Candidate materials for LFR/ QuaDeSIM/, Marc Vankeerberghen and Serguei Gavrillov, SCK CEN

The presentation "QuaDeSIM - Overview" by Marc Vankeerberghen and Serguei Gavrillov provides a detailed account of the QuaDeSIM project which is focused on the Qualification and Design of Experiments for Structural Integrity of Materials in lead (Pb) and lead-bismuth eutectic (LBE) environments. This project (2023 – 2027) is aimed at enhancing the understanding and methodologies around the use of these heavy liquid metals as coolants in nuclear reactors particularly those classified under Generation IV Small Modular Reactors (SMRs) and transmutation systems.

The necessity for this project arises from the increasing complexity and stringent safety requirements of modern nuclear systems which require highly reliable materials that can withstand aggressive operating conditions. The primary goal of QuaDeSIM is to develop a systematic approach for generating reliable and structured data that can be used for the accurate assessment of material behavior under these conditions. This involves a rigorous process of experimentation and data analysis, focusing on mechanical and corrosion properties of structural materials.

During the project various methodologies will be evaluated and optimized, including design of experiments (DOE), which is crucial for determining the key factors affecting material integrity. Data gathered from these experiments will be managed through a

sophisticated platform, MatDB, ensuring that all findings are documented and accessible for future research and application. This platform will also facilitate the validation and quality assessment of the data by expert work package leaders.

QuaDeSIM aims to fill the existing knowledge gaps in material science for nuclear applications by conducting targeted experiments and developing best practice guidelines. These efforts are expected to lead to improved material qualification processes, which are critical for the licensing and operation of next-generation nuclear reactors.

By the end of the project QuaDeSIM will provide a comprehensive framework for the qualification of materials used in advanced nuclear systems. This will not only support the design and safety of new reactors but also contribute to the overall sustainability and efficiency of nuclear energy systems. The project outcomes are anticipated to be a significant step forward in the standardization of testing and evaluation protocols for nuclear materials impacting future developments in the field.

2.2.5 CNL activities on VHTR/HTGR candidate materials, Cathy Thiriet, CNL

The presentation "CNL Activities on VHTR/HTGR Candidate Materials" by Cathy Thiriet at Canadian Nuclear Laboratories delves deeper into the research and development of candidate materials suitable for high-temperature gas-cooled reactors (HTGR) and very high temperature reactors (VHTR). The presentation outlines CNL efforts to meet the challenges posed by the need for materials that can withstand high temperatures and corrosive environments which are critical to the success of next-generation nuclear reactors. The work focuses on a variety of materials including advanced steels and alloys like Alloy 800H, a potential material candidate for in-core components. This alloy must be further qualified at temperatures higher than 760 °C, the current qualification temperature upper limit. In addition, finding a suitable filler material employed for Alloy 800H weldments is critical to the applicability range of this promising candidate material to higher temperatures. High-temperature creep testing under different stress conditions is ongoing to define alternative filler materials for Alloy 800H weldments based on its performance in terms of creep resistance, mechanical strength and corrosion resistance.

The research aims to provide crucial data to the designers and operators of HTGRs and VHTRs supporting their role in achieving Canada's net-zero emissions targets. By improving the understanding of how these materials perform under extreme conditions CNL contributes to the global effort to make nuclear energy safer and more efficient. Additionally, the presentation discusses the collaboration with other institutions and the integration of findings into international standards which helps in advancing the nuclear materials field globally. This is part of Canada's broader strategy to enhance its energy infrastructure and reduce its reliance on fossil fuels emphasizing the role of nuclear technology in a diversified energy future.

2.2.6 Molten Salt Chemistry Capabilities at Canadian Nuclear Laboratories (CNL), Liyan Qiu, CNL

The presentation "Molten Salt Chemistry Capabilities at CNL" by Liyan Qiu provides an extensive overview of Canadian Nuclear Laboratories research related to molten salt chemistries for advanced nuclear reactors specifically molten salt reactors (MSRs). It

highlights the unique challenges associated with MSR's such as their operation at high temperatures around 700°C, excellent thermal conductivity and issues related to the hygroscopic nature and purity of molten salts which can lead to severe structural material corrosion.

The presentation delves into CNL comprehensive capabilities in addressing these challenges including developments in salt purification techniques, corrosion management and the electrochemistry of molten salts. Significant attention is given to the improvement of methodologies for analyzing thermophysical properties and the solubility and transport of corrosion products which are critical for the integrity and longevity of reactor materials.

Additionally, it outlines CNL efforts in advancing molten salt chemistry through experimental setups like Differential Scanning Calorimetry/Thermogravimetric Analysis (DSC/TGA) and the use of inert atmospheres for precise measurements. The research also encompasses the study of fission product chemistry within molten salts which has substantial implications for waste management and reactor safety.

By enhancing the understanding of molten salt behavior and its interaction with structural materials CNL aims to provide data that supports the safe and efficient operation of future Generation IV nuclear reactors. This work not only contributes to the field of nuclear materials but also aligns with global energy goals by facilitating the development of clean and sustainable nuclear energy solutions.

2.2.7 Atmospheric Corrosion of Additively Manufactured Stainless Steel 316L for Nuclear Canister Applications, Tasnia Fatima, Brendy Rincon Troconis, the University of Texas at San Antonio (UTSA)

The presentation "Atmospheric Corrosion of Additively Manufactured Stainless Steel 316L for Nuclear Canister Applications" by Brendy Rincon Troconis outlines metal additive manufacturing (AM) methods, such as selective laser melting (SLM), which have attracted considerable attention within the manufacturing and materials science community. This technology allows for layer-by-layer fabrication of metallic parts, offering numerous advantages. Metal AM enables the production of parts at the point of need, reducing logistical challenges and costs associated with traditional manufacturing methods. SLM, specifically, is a powder-based technique where a layer of metal powder is selectively melted and solidified to form the desired geometry.

Carbon nanotubes (CNTs) have attracted considerable attention as potential reinforcements for metal matrix composites due to their exceptional mechanical properties. CNT containing materials have shown enhanced hardness and wear resistance of the composite in comparison to unreinforced SLM fabricated 316L SS materials. Also a recent study reported that the as received milled SS316L specimen exhibited low pitting susceptibility compared to SLM SS after conducting electrochemical test under immersion conditions. Nevertheless, there is no information on the corrosion performance of this novel material under atmospheric conditions.

The objective of this research is to study the atmospheric pitting corrosion resistance of SLM 316L fabricated from CNT containing feedstock under MgCl₂ solution. In this research four types of materials were used, including conventional SS316L, non-CNT containing SLM SS316L, and two CNT integrated SLM SS316L (1 and 2 %CNT). Samples loaded with MgCl₂ salt solution droplets with same initial chloride deposit density and volume were subjected to 40% RH at room temperature (23°C) exposure during 30 days. After exposure, the corrosion morphology was characterized using optical profilometry and scanning electron microscopy. SLM printed samples showed lower pit

depth, volume, area, and diameter than the as received SS316L. Also, different pit morphologies were observed for as received and SLM samples. As received samples showed crystallographic faceted pit morphology while the SLM samples showed equiaxed and columnar cellular structure pitting.

The results from this research demonstrate the feasibility of advancing materials performance via additive manufacturing and modification of feedstock material, leading to the potential use of LM 316L and CNT-containing SLM 316L materials in different applications, including storage of spent nuclear fuels.

2.3 SESSION 3: INFLUENCE OF ENVIRONMENT, COOLANT AND RADIATION ON STRUCTURAL AND CLADDING MATERIALS FOR SCW SMR

2.3.1 Fuel Cladding: an introduction, Alberto Sáez-Maderuelo, CIEMAT

The presentation "Cladding: an Introduction" by Alberto Sáez-Maderuelo expands on the role of cladding materials in ensuring the safety and efficiency of nuclear reactors, particularly focusing on the newer technologies like Supercritical Water Reactors (SCWRs) and Small Modular Reactors (SMRs). It discusses the historical context and evolution of nuclear power and emphasizes the shift from military to civilian energy uses and the technological advancements in reactor design.

The presentation explains the critical attributes required for cladding materials such as mechanical durability at high temperatures, corrosion resistance and the ability to withstand neutron irradiation without significant degradation. These properties are essential for maintaining the integrity of the reactor core and by extension, the safety of the nuclear plant. Cladding acts as a barrier that keeps radioactive fission products contained within the fuel rods and prevents them from contaminating the reactor coolant.

Moreover, it addresses the selection process for cladding materials which involves an iterative testing and evaluation method. This process ensures that the materials chosen can perform under the demanding conditions of SCWRs and SMRs which operate at higher temperatures and pressures than traditional reactors, thus posing a unique set of challenges. The ECC-SMART project, as part of its materials testing work package 2, specifically focuses on evaluating the corrosion behavior of candidate alloys for cladding in supercritical water conditions.

The presentation highlights the importance of international collaboration in advancing nuclear technology, as demonstrated by the joint efforts of European, Canadian and Chinese researchers within the ECC-SMART project. This collaborative approach not only accelerates the development of safe and efficient nuclear reactors but also fosters the sharing of knowledge and resources across borders.

In conclusion "Cladding: an Introduction" underscores the critical role of material science in the next generation of nuclear reactors showcasing how advancements in cladding materials contribute to the broader goals of safety, efficiency and sustainability in the nuclear energy sector.

2.3.2 Oxidation of SCW-SMR Candidate Cladding Materials at CNL, K. Khumsa-Ang, L. Yao, M. Edwards, H. Namburi and A. Nava-Dominguez, CNL

The presentation "Oxidation of SCW-SMR Candidate Cladding Materials at CNL" by K. Khumsa-Ang and colleagues delves deeper into the corrosion challenges faced by cladding materials used in the construction of Supercritical Water-cooled Small Modular Reactors (SCW-SMRs). The team at Canadian Nuclear Laboratories conducted extensive testing to simulate operational conditions of SCW-SMRs focusing on high temperatures and pressures. These experiments were designed to evaluate the long-term stability and durability of materials such as Alloy 800H, 310S and Alumina-forming Alloys (AFAs).

The research methodology included a series of oxidation tests conducted at 380°C and 500°C with the materials subjected to these conditions for durations extending up to 4000 hours to mimic the harsh operating environments of SCW-SMRs. The objective of the study was to capture detailed data on the oxidation kinetics and mechanisms that affect these materials under supercritical water conditions. Post-experimental analysis was comprehensive and involved morphological studies of the oxides formed, elemental analysis of both the oxide layers and the underlying materials and assessments of the physical changes in weight and structural integrity.

One key aspect of the research was the focus on the elemental composition and the structural evolution of the oxidation layers which are crucial for understanding how these materials degrade and how their degradation impacts the overall safety and efficiency of the reactor. The experiments utilized advanced techniques such as Energy Dispersive X-Ray (EDX) analysis to provide a detailed understanding of the changes occurring on a microscopic scale.

This ongoing research is critical for advancing the knowledge base around the materials science of nuclear reactors, particularly those that operate under the challenging conditions proposed for next-generation nuclear technologies. The findings from CNL studies are expected to contribute significantly to the design and development of more robust and reliable cladding materials ensuring the long-term operational safety and viability of SCW-SMRs. This research not only supports the development of new reactor designs but also provides valuable insights that could lead to improvements in existing technologies.

2.3.3 Corrosion and Stress Corrosion Cracking in SCW: results from ECC-SMART H2020 Project, R. Novotny, EC Joint Research Centre

The presentation "Corrosion and Stress Corrosion Cracking in SCW: results from ECC-SMART H2020 Project" by R. Novotny provides a comprehensive examination of the impact of supercritical water (SCW) conditions on the degradation of materials used in nuclear reactors. It focuses on the behavior of stainless steel 310S and alloys such as Alloy 800H and AFA under these conditions. The study is crucial for understanding how these materials withstand the combination of high temperatures and pressures found in SCWRs, one of the types of Generation IV nuclear reactors.

The research encompasses various testing methodologies including oxidation tests and Slow Strain Rate Tensile (SSRT) tests which are designed to simulate the conditions these materials would face in reactors in operation. This includes examining the

resistance to corrosion, the ability to maintain mechanical integrity under stress and the overall durability of the materials over extended periods. Results from these tests are critical in determining the viability of these materials for long-term use in nuclear reactors and thus proof reactor safety.

Additionally, the presentation delves into the specifics of the experimental setups used in the ECC-SMART H2020 project highlighting the sophisticated techniques employed to assess material performance. This includes detailed analyses of the corrosion processes and the mechanisms of stress corrosion cracking providing insights into how different materials react under various conditions of stress and temperature.

The findings of this research are intended to guide the selection of materials for future SCWRs by identifying those that offer the best combination of corrosion resistance and mechanical strength. This work is part of a broader effort to enhance the safety and efficiency of next-generation nuclear reactors contributing to the development of more sustainable and cost-effective nuclear power options. This type of research is vital for advancing our understanding of material science in nuclear applications and pushing forward the boundaries of what is possible in nuclear reactor design.

2.3.4 Effect of Neutron Irradiation on Mechanical Properties of Cladding and Structural Materials, Monika Šípová, CVR

The presentation "Effect of Neutron Irradiation on Mechanical Properties of Cladding and Structural Materials" by Monika Šípová (Ref.: Fig.3) offers an in-depth analysis of how neutron radiation impacts the integrity of materials used in nuclear reactors. Implemented as a part of the ECC-SMART project, the study examines the changes caused by neutrons and their effect on mechanical properties. The presentation details experimental methods including tensile tests conducted at room temperature and at 500 °C (corresponding to the expected outlet temperature of SCW-SMR), nanoindentation techniques and microstructural analysis using Transmission Electron Microscopy (TEM).

The presentation also contains a brief introduction to the main degradation phenomena caused by neutron radiation including radiation hardening, embrittlement and material swelling due to void formation. These effects are critical to understanding as they influence the safety and operational lifetime of nuclear reactors. The research highlights the importance of selecting and developing materials that can withstand such challenging environments with a particular focus on the long-term stability and reliability of cladding and structural materials.

Further, the presentation covers the role of neutron energy spectrum and flux on the severity of radiation damage providing insights into how different materials react to varied radiation exposures. This is important for designing materials that can endure the harsh conditions inside new generation nuclear reactors while maintaining their structural integrity and mechanical properties. The ECC-SMART project findings are intended to guide improvements in material science for nuclear applications and ensure advancements in safety and technology in the nuclear energy sector.



Fig.3: Effect of Neutron Irradiation on Mechanical Properties of Cladding and Structural Materials, Monika Šípová, CVR; Source: ECC-SMART Project

2.3.5 Materials for Supercritical CO₂ Nuclear Power Generation Systems, L. Zhang, Shanghai Jiao Tong University, China

The presentation "Materials for Supercritical CO₂ Nuclear Power Generation Systems" by Le-Fu Zhang delves deeper into the application of supercritical CO₂ (SC-CO₂) in nuclear systems emphasizing the selection and testing of materials under extreme conditions. It provides a comprehensive look at the unique properties of SC-CO₂ such as its efficiency at high temperatures and pressures which makes it an ideal working fluid for modern nuclear reactors. The study evaluates the corrosion, creep and cracking challenges faced by materials like austenitic stainless steels and novel alloy formulations in high temperature SC-CO₂ environments.

This research also addresses specific phenomena such as oxidation, carburization and stress corrosion cracking that occur in these environments. It discusses the mechanisms and effects on material integrity. The presentation outlines ongoing research and development efforts to optimize alloys by enhancing chromium content and integrating elements like aluminum and silicon to improve performance against corrosion and mechanical stress.

Further, it highlights experimental studies on commercial and novel materials and assesses their performance in resisting environmental degradation. The research aims to bridge the gap between current material capabilities and the demanding requirements of SC-CO₂ nuclear systems pushing forward the development of more durable and reliable materials.

2.3.6 In-situ Electrochemical Corrosion Testing of Candidate Materials for SCWR Fuel Cladding, Jan Macák, David Dašek, Petr Roztočil, Petr Čech, Jaromír Valtr, Mariana Arnoult-Růžicková, Petr Sajdl, University of Chemistry and Technology, Prague

The presentation "In-situ Electrochemical Corrosion Testing of Candidate Materials for SCWR Fuel Cladding" by David Dašek and colleagues from the University of Chemistry and Technology Prague focuses on the use of in-situ electrochemical methods to evaluate corrosion in supercritical water (SCW) environments. This innovative approach compared to traditional ex-situ methods aims to provide more reliable and immediate corrosion data for materials used in SCWRs (Supercritical Water-Cooled Reactors). The presentation details the experimental setup and methodology for electrochemical impedance spectroscopy (EIS) measurements across critical temperature transitions and discusses the interpretation of these measurements to assess the corrosion characteristics of various candidate materials at elevated temperatures and pressures.

2.3.7 Wall Effects on Turbulent SC Flow: Results from ECC-SMART Project, Ivan Otic, KIT Germany

The presentation "Wall Effects on Turbulent SC Flow: Results from ECC-SMART Project" by Ivan Otic focuses on the impact of wall properties on the turbulent supercritical (SC) fluid flow specifically within the context of nuclear reactor simulations and applications.

It delves into the use of Conjugate Heat Transfer (CHT) simulations and Direct Numerical Simulations (DNS) to analyze fluid-solid interactions under varying wall thickness conditions in SC environments. It also covers experimental setups and results emphasizing the significance of accurate simulation models in predicting real-world reactor behavior. Additionally, the presentation incorporates machine learning techniques to enhance the prediction and analysis of these complex fluid dynamics representing a cutting-edge integration of computational fluid dynamics and artificial intelligence in nuclear reactor design and safety analysis.



Fig.4: Wall Effects on Turbulent SC Flow: Results from ECC-SMART Project, Ivan Otic, KIT Germany; Source: ECC-SMART Project

2.3.8 Accelerated High Energy Proton Irradiations and Small-Scale Mechanical Testing of Alloy 800H and 310 SS Candidate Fuel Cladding Materials for SCWR Applications, H. Namburi & L.Walters, CNL

The presentation "Accelerated High Energy Proton Irradiations and Small-Scale Mechanical Testing of Alloy 800H and 310 SS Candidate Fuel Cladding Materials for SCWR Applications" by H. Namburi and Lori Walters provides an in-depth look at the effects of high energy proton irradiations on materials intended for use in Supercritical Water Reactors (SCWRs). The study focuses on two key materials Alloy 800H and 310 SS assessing their mechanical properties and resistance to irradiation.

The presentation details the use of accelerated irradiation experiments to simulate neutron radiation damage and includes small-scale mechanical testing to evaluate the performance of these materials under simulated reactor conditions. This research is part of the ECC-SMART project aimed at advancing the understanding and development of materials capable of withstanding the harsh environments of SCWRs.

2.4 SESSION 4: ADVANCED MICROSTRUCTURE EXAMINATION AND ADVANCED TESTING METHODS

2.4.1 Advanced Microstructure Examination Techniques in Support of Gen IV SMRs Materials Research, Daniela Marušáková, Patricie Halodová, CVR

The presentation "Advanced Microstructure Examination Techniques in Support of Gen IV SMRs Materials Research" by Daniela Marušáková and Patricie Halodová from Czech Research Centre Řež highlights the challenging conditions such as high temperatures, neutron radiation and corrosive environments that Gen IV SMRs will operate under. It emphasizes the need for materials that do not only keep high strength but also exhibit superior corrosion resistance.

The researchers outline the use of several advanced microstructural characterization techniques to understand and enhance the material development. Techniques such as Transmission Electron Microscopy (TEM), Scanning Electron Microscopy (SEM), Scanning Probe Microscopy (SPM) and Atomic Force Microscopy (AFM) are discussed in detail. These methods allow for a detailed analysis of material degradation mechanisms connected with e.g. creep, corrosion fatigue and environmentally assisted cracking.

Furthermore, the presentation covers the application of state-of-the-art analytical tools like Energy Dispersive Spectrometry (EDS), Wavelength Dispersive Spectrometry (WDS) and Electron Energy Loss Spectroscopy (EELS) to investigate the changes in materials at almost the atomic level. This comprehensive suite of techniques provides insights into how materials behave under stress and radiation studying potential for the development of new materials with tailored properties to meet the specific demands of next generation nuclear reactors.

By employing these sophisticated examination techniques, the research group aims to create a robust database of material behaviors which will provide this information to the designers and manufacturers of new generation reactors. The presentation not only underscores the importance of material characterization in the lifecycle of nuclear

reactors but also highlights the collaborative effort required to advance the field of nuclear materials science which is crucial for the success of Gen IV SMRs.



Fig.5: Advanced Microstructure Examination Techniques in Support of Gen IV SMRs Materials Research, Daniela Marušáková, CVR; Source: ECC-SMART Project

2.4.2 Residual Stress Influence on Cracking of 316L Cold Spray (Genetic Algorithms), T. De Terris (EDF), A. Saez and Francisco J. Perosanz (CIEMAT), Ricardo Fernández and Gaspar González (CSIC)

The presentation "Residual Stress Influence on Cracking of 316L Cold Spray (Genetic Algorithms)" by Ricardo Fernández and colleagues explores the impact of residual stresses on stress corrosion cracking (SCC) in 316L stainless steel applied via cold spray on a 304 substrate.

This study uses advanced techniques like Electron Backscatter Diffraction (EBSD) and X-ray diffraction to assess the microstructural characteristics and residual stress distributions within the cold-sprayed material. Genetic algorithms are proposed to analyze data focusing on the interaction between microstructural features and the development of stress-induced cracks.

The research findings aim to enhance the understanding of SCC mechanisms in cold-sprayed coatings crucial for improving material performance in demanding environments.

2.4.3 Advanced and Conventional Electrochemical Methods for Corrosion Evaluation, Jesús M. Vega, Universidad Complutense de Madrid

The presentation "Advanced and Conventional Electrochemical Methods for Corrosion Evaluation" by Jesús M. Vega from Universidad Complutense de Madrid delves into the effectiveness of both advanced and conventional electrochemical techniques in assessing corrosion mechanisms.

It emphasizes the adaptation of these methods to the unique demands of GEN IV SMR Concepts, focusing on their application in evaluating magnesium, aluminum, titanium, zinc and zirconium alloys. The presentation covers the protective capabilities of various surface treatments such as anodizing and Plasma Electrolytic Oxidation (PEO) and explores their impact on material degradation under aggressive conditions.

The study integrates detailed characterization tools like SEM, TEM and AFM to provide insights into the microstructural changes that occur during corrosion processes.

2.5 SESSION 5: DEVELOPMENT OF ADVANCED MATERIALS SOLUTIONS, FABRICATION PROCESSING AND OTHER TOPICS

2.5.1 Reduced oxygen content of PM HIP materials for nuclear power plants, I. Heikkilä, et al., SWERIM

The presentation "Reduced Oxygen Content of PM HIP Materials for Nuclear Power Plants" by I. Heikkilä and colleagues focuses on the benefits of using Powder Metallurgy Hot Isostatic Pressing (PM HIP) to produce materials with reduced oxygen content for nuclear power applications.

The presentation details the investigative process aimed at enhancing the toughness of PM HIP materials like SA-508 and SS-316L by reducing the oxygen impurities that negatively impact material performance. It highlights the integration of an oxygen reduction treatment in the manufacturing process which shows significant improvements in material properties such as toughness and corrosion resistance.

The results demonstrate potential enhancements in the safety and efficiency of materials used in nuclear power plants, making a strong case for the adoption of PM HIP techniques in the industry.

2.5.2 Design Code Evolution and Accelerated Qualification of Innovative Nuclear Reactor materials & components, Karl-Fredrik Nilsson, EC Joint Research Centre

The presentation "Design Code Evolution and Accelerated Qualification of Innovative Nuclear Reactor Materials & Components" by Karl-Fredrik Nilsson addresses the need for a paradigm shift in the materials and component qualification for nuclear reactors.

Focused on enhancing safety, reducing costs and accommodating new operational conditions such as higher temperatures and radiation levels, this presentation delves into the evolution of design codes and qualification approaches. It highlights the integration of advanced manufacturing methods like 316L stainless steel additive manufacturing and new material classes such as Aluminium Forming Austenitic (AFA) steels.

The presentation covers the role of the European Commission DG-JRC in pushing forward these advancements through initiatives like CEN WS064 – Code Evolution Project, aiming to streamline and expedite the qualification process for nuclear materials and components.

2.5.3 State-of-the-art of ATF Research and Lessons Learned for Gen IV SMR, Martin Sevecek, Czech Technical University in Prague (CTU)

The presentation "State-of-the-art of ATF Research and Lessons Learned for Gen IV SMR" by Martin Sevecek offers a comprehensive review of the developments and challenges associated with Accident Tolerant Fuels (ATFs) for nuclear reactors. It traces the historical evolution from early nuclear submarine technologies to the current emphasis on enhancing fuel performance to withstand severe accidents better.

It discusses the progression of ATF technologies including the integration of new materials like FeCrAl alloys and SiC/SiC composites. The presentation addresses the motivations behind ATF development, especially in response to failures observed in past reactor incidents and evaluates the impact of these technologies on reactor safety and efficiency.

Emphasizing the need for accelerated testing and qualification processes, the presentation highlights the significant advancements in materials science that can potentially lead to safer and more reliable nuclear power generation.

2.5.4 The Development of New Grade Heat-Resistant Iron Based Alloys with Improved Corrosion Resistance, Zhangjian Zhou, University of Science & Technology Beijing

The presentation "The Development of New Grade Heat-Resistant Iron-Based Alloys with Improved Corrosion Resistance" by Zhangjian Zhou from the University of Science & Technology Beijing focuses on the creation and evaluation of advanced metallic materials for Generation IV Small Modular Reactors (SMRs). It particularly examines Alumina Forming Austenitic (AFA) steels and Oxide Dispersion Strengthened (ODS) steels, detailing their high-temperature mechanical properties, corrosion resistance in harsh environments like supercritical water and liquid lead and their low susceptibility to stress corrosion cracking.

It also discusses the specific microstructural features that enhance the performance of these materials. For AFA steels, the focus is on their ability to form stable passivating layers of alumina which significantly improves their oxidation resistance at high temperatures. In the case of ODS steels, the presence of finely dispersed yttria particles within the steel matrix contributes to their superior high-temperature strength and radiation damage resistance, critical for their application in nuclear reactor environments.

The research emphasizes the importance of these materials in enhancing the operational efficiency and safety of future nuclear reactors, addressing both their fabrication technologies and cost-effectiveness in the nuclear industry.

2.5.5 FAIR Data Management Services in Support of the European Nuclear Energy Technology Community, Tim Austin, EC Joint Research Centre

The presentation "FAIR Data Management Services in Support of the European Nuclear Energy Technology Community" by Tim Austin outlines the efforts of the European Commission's Joint Research Centre (JRC) to implement FAIR (Findable, Accessible, Interoperable, Reusable) data management principles within the nuclear energy sector.

It discusses the MatDB database which supports engineering materials data management for Euratom projects, enhancing data quality, citation and standards. The presentation emphasizes the significance of data sharing and reuse in accelerating the development and licensing of advanced nuclear technologies such as Liquid Metal-cooled reactors and highlights ongoing international collaborations facilitated by these data practices.

2.5.6 Mitigating Environmentally-Assisted Cracking Through Optimisation of Surface Condition, F. J. Perosanz, CIEMAT

The presentation "Mitigating Environmentally-Assisted Cracking Through Optimisation of Surface Condition" by F. J. Perosanz and colleagues based on findings of the MEACTOS Euratom project and focuses on innovative machining techniques to improve the resistance of austenitic alloys to environmentally assisted cracking (EAC) in light water reactors (LWR).

It discusses the application of advanced surface treatments such as Minimum Quantity of Lubricant (MQL) and cryogenic machining which have shown potential in reducing tensile residual stresses and enhancing surface integrity. The study also examines the integration of cryogenic machining and MQL processes, emphasizing their ability to reduce the environmental impact and improve machining efficiency. These processes are scrutinized through experimental approaches and testing matrices that include varied alloys and surface conditions across different reactor environments. This comprehensive testing aims to validate the efficacy of the machining treatments in significantly reducing crack initiation risks in nuclear materials.

These methods along with the use of supercritical CO₂ as a cutting fluid have been evaluated for their environmental impact, cost savings and improvement in tool life and productivity. The presentation highlights the benefits of these machining techniques in the nuclear sector, particularly in enhancing the durability and performance of materials under corrosive conditions commonly encountered in nuclear reactors.

2.5.7 ENEN & What Can We Offer to Students, Roberta Cirillo, Gabriel Pavel, Kateryna Piliuhina, ENEN

The presentation "Building European Nuclear Competence Through Continuous Advanced and Structured Education and Training Actions" by Roberta Cirillo, Gabriel Pavel and Kateryna Piliuhina focuses on the European Nuclear Education Network (ENEN) efforts to enhance nuclear competence across Europe.

The presentation outlines ENEN role in promoting advanced and structured education and training actions to support nuclear professionals. It details ENEN activities, including managing the largest education and training network in Europe promoting collaboration in nuclear education and ensuring quality and engagement in the nuclear field.

It also emphasizes ENEN role in facilitating mobility, supporting career development and coordinating participation in EU-funded projects like ENEN2plus Euratom project. It highlights how ENEN efforts contribute to the development and preservation of nuclear expertise, addressing both immediate educational needs and long-term industry challenges.

3 Involvement of the students

This workshop, as well as the previous one on the Safety of Small Modular, Advanced and Fusion Reactors held in Slovenia in June 2023, also contributed to the ECC-SMART project dedication to education and training.

The collaboration with the ENEN2plus project has fostered education and training improvements and career development by offering its support to students.

10 students from various parts of Europe participated in the activities held during the Workshop (Ref.: Fig.6). Most are currently working on their MSc or PhD theses which focus on areas such as mechanical engineering, materials studies, thermal-hydraulic research, innovative nuclear technologies and the licensing of nuclear reactors. They all intend to contribute their research to future SMR (Small Modular Reactor) projects. A few students also presented their work and experience related to material research for GenIV. The reports on the selection process compiled as part of the ENEN2plus project requirements can be found in Annex A, appendix 3.

The workshop has laid the groundwork for ongoing and future project endeavors. It was announced that similar collaborative and educational efforts will continue throughout the remaining stages of the project.



Fig.6: Students at the Workshop on the Structural Materials for Gen IV SMR Concepts, November 2023, Madrid, Spain.

4 Conclusions

The ECC-SMART project Deliverable 6.2 "Proceedings of Workshop on Materials and Online In-Situ Methods for Corrosion Monitoring in SCWR" serves as a comprehensive account of the discussions and findings from the workshop held in Madrid in November 2023. The document synthesizes the expert contributions from various nuclear safety and technology fields focusing on advancements in structural materials for Generation IV Small Modular Reactors and the innovative techniques developed to monitor and evaluate material corrosion in supercritical water reactors.

The workshop provided an important professional platform for sharing research insights, fostering international collaboration and driving forward the technical and safety standards necessary for the future of nuclear energy development. This summary encapsulates the collective efforts and scientific progress presented, highlighting the workshop's role in enhancing understanding and addressing the challenges facing the nuclear energy sector.

The achievements of the workshop play an important role in addressing the dissemination objectives of the ECC-SMART project. It establishes a foundation for ongoing collaborations and educational endeavors, guaranteeing the lasting influence of the project and recognition even after its conclusion.

Annex_A

All below-mentioned appendices are available to the project consortium on the ECC-SMART SharePoint.

Annex_A – List of appendices

Number of app.	Title of the appendix	Note
1	Workshop Programme	Available on the ECC-SMART SharePoint
2	Presentations	Available on the ECC-SMART SharePoint
3	Involvement of Students	Available on the ECC-SMART SharePoint