Croatian Nuclear Society

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EUROPEAN NUCLEAR SOCIETY

FER

15th International Conference of the

CROATIAN NUCLEAR SOCIETY

Nuclear Option for Reliable Electricity Generation

May 31 – June 3, 2026, Zadar, Croatia

Book of Abstracts

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Conference Organized by

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in cooperation with International Atomic Energy Agency,
European Nuclear Society, and University of Zagreb
Faculty of Electrical Engineering and Computing

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Foreword

Croatian Nuclear Society continues with its successful series of international conferences for professionals working in the field of nuclear energy and associated areas. This International Conference of the Croatian Nuclear Society (HND2026), subtitled "*Nuclear Option for Reliable Electricity Generation*", is already the 15th event in the successful series of international conferences, formerly known as "*Nuclear Option in Countries with Small and Medium Electricity Grids*", biennially organized by the Croatian Nuclear Society.

The purpose of conference series is to present and discuss the most relevant topics concerning the role and position of nuclear option in the current energy balance, with special attention paid to the countries with small and medium electricity grids. Main concerns with which modern society is faced include availability of energy resources, resilience of electrical energy systems, greenhouse gas emissions and potential climate changes with evidence of apparent global warming. In such a context the issue of ensuring reliable and sustainable energy becomes ever more challenging.

Expansion of nuclear energy can provide a robust, low carbon foundation for reliable electricity generation, but the benefits depend on careful technology selection, grid adaptation, outage management, supply chain assurance and regulatory preparedness. Small modular reactors (SMRs) are well matched to smaller electrical systems by reducing single unit risks and enabling phased investment. Successful deployment requires integrated technical, economic and institutional planning, and, where appropriate, regional cooperation on fuel services and waste management to maintain continuous, secure electricity supply.

Nuclear option thus offers a proven technical path to reliable, low carbon electricity, recognized with high-capacity factors, predictable output, and long operational lifetime. It can make an effective backbone for modern, decarbonized power systems. When appropriately sized and integrated, nuclear option reduces exposure to fuel price volatility, supports system adequacy, and complements variable renewables and storage by providing firm, dispatchable capacity. Successful deployment requires coordinated planning across technical, regulatory, and financial domains and, where appropriate, regional cooperation to manage backend services and grid reliability. In sum, nuclear option can be a dependable cornerstone of a resilient, decarbonized electricity system if technical, institutional and financial frameworks are established up front to manage construction, operation, safety, and the back-end lifecycle.

Following the success of the previous conferences in the series, the 15th International Conference in Zadar serves the same general purpose, concentrating on the topics which attracted the most of interest previously. At the Conference, the nuclear option is considered and discussed from the point of view of national energy strategies, resilience, resources, costs, and technological, organizational, and educational requirements, as well as environmental advantages. The focus is on matters related to nuclear power plants operation and design safety, fuel cycle, waste management, decommissioning, and achievement of

Long-Term Operation (LTO). As in the previous cases, the important goal of the Conference is to promote regional co-operation and exchange of experience in use of nuclear power and fuel cycle facilities among the countries with an interest in the nuclear option.

Authors' and presenters' contributions are provided in 11 invited lectures and almost 100 papers have been contributed in total. The contributed papers are grouped into eight thematic sessions:

- S1: Nuclear Safety Analyses (NSA)
- S2: Operation, Maintenance and Lifetime Expansion Experience (OMLEE)
- S3: Nuclear Option in the Context of Energy, Economics, Finance and Resilience (NEEFR)
- S4: Regulatory Practice, Licensing, Emergency Preparedness, Safety Culture and Public Relations (RPLEP)
- S5: Reactor Physics and Nuclear Fuel Cycle (RPNFC)
- S6: Severe Accident Analyses and Risk Assessment (SAARA)
- S7: Radioactive Waste Management and Decommissioning, Radiation Hazard and Protection (RWMD)
- S8: Small Modular Reactors (SMRs)

A special topic to this HND2026 conference is implementation (design, licensing and operation) of small modular reactors (SMRs) which is a subject of several invited presentations. This topic is additionally addressed under the round table discussion, which is one of the focal points of the conference. SMRs with typical electric power output up to 300 MWe are designed for factory fabrication and modular on-site assembly. They offer lower single unit risk for small electrical grids, shorter construction times, and staged capacity additions. Key benefits can be seen in enhanced siting flexibility, potential cost reductions via standardization, and suitability for pairing with renewables. However, their successful development path is facing several challenges that need to be resolved, like licensing maturity, supply chain scale up, financing models, and spent-fuel management.

This "Book of Abstracts" provides printed abstracts of contributed papers, invited presentations and summarized outlines and topics for the round table discussion. The Proceedings with peer-reviewed full papers will be distributed on USB memory key during the Conference.

We would like to express our gratitude to nearly 200 authors and co-authors that put a large effort into completing 87 full camera-ready papers. We would also like to thank the sessions' coordinators and chairs, reviewers, and all those who gave a hand in organizing the HND2026 Conference.

Special acknowledgments are given to the International Atomic Energy Agency, the European Nuclear Society, and University of Zagreb Faculty of Electrical Engineering and Computing for their support.

Finally, we are particularly grateful to all the sponsors and donors whose help has been essential for the success of this International Conference. We express our thanks to all those who, through their efforts and participation, have contributed to the Conference's success.

Zagreb, May 2026

EDITORS

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Round Table (RT)

SMRs today and tomorrow

Round Table Discussion will be moderated by: DAVOR GRGIĆ

Invited speakers will attend round table and discuss the following subjects:

- **When are you expecting to see first operating western SMR and where?**
- **What is an optimum installed power of SMR module and why? What are the main drivers?**
- **What is an acceptable increase of SMR price per kWe compared to a large LWR?**
- **What to do to advance first of a kind to n-th of a kind in an efficient way?**
- **Is it possible to change soon regulatory requirements and reduce EPZ to EAB for SMRs?**
- **What do you think about passive systems as a first line of defence in SMR nuclear safety?**
- **What do you think about SMR and AMR common future?**

Session 1 *Nuclear Safety Analyses (NSA)*

Session 2 *Operation, Maintenance and Lifetime Expansion Experience (OMLEE)*

Session 3 *Nuclear Option in the Context of Energy, Economics, Finance and Resilience (NEEFR)*

Session 4 *Regulatory Practice, Licensing, Emergency Preparedness, Safety Culture and Public Relations (RPLEP)*

Session 5 *Reactor Physics and Nuclear Fuel Cycle (RPNFC)*

Session 6 *Severe Accident Analyses and Risk Assessment (SAARA)*

Session 7 *Radioactive Waste Management and Decommissioning, Radiation Hazard and Protection (RWMD)*

Invited Presentations (IP)

IP-1

KONČAR's Long-Term Operational and Maintenance Support at the Krško Nuclear Power Plant

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KONČAR – Electrical Industry Inc. has maintained an enduring presence in the nuclear energy sector for more than 40 years, with long-term involvement in maintenance and service activities at the Krško Nuclear Power Plant (NEK). Our cooperation focuses on delivering preventive and corrective maintenance activities on key electrical machinery that underpin the reliability and operational readiness of one of the region's most important low-carbon power generation facilities.

Within this partnership, KONČAR has been engaged in the regular maintenance of NEK's main generator, two auxiliary diesel generators, and 36 high-voltage motors according to the plant's 18-month outage schedule. These activities are integral to supporting safe, dependable operation within a highly regulated nuclear environment.

KONČAR's scope of work encompasses scheduled inspections, comprehensive diagnostics, mechanical and electrical refurbishment, component repair and replacement, performance testing and verification in accordance with applicable technical and quality assurance standards. Activities are executed both on-site during planned outages and supported by specialized workshop capabilities, ensuring traceability, rigorous documentation and compliance with safety requirements. Through decades of cooperation, KONČAR has built in-depth technical knowledge of maintenance of large rotating electrical machines operating in nuclear power applications, as well as robust processes for outage planning, rapid fault response and long-term asset management.

In addition to site-specific activities at NEK, KONČAR's experience in the nuclear sector extends globally. To date, the company has delivered 78 high-voltage synchronous generators and asynchronous motors to nuclear power plants worldwide, demonstrating its capability to meet stringent performance and quality requirements for critical power generation equipment.

This presentation will provide an overview of KONČAR's operational and maintenance framework at the Krško Nuclear Power Plant, highlight practical experience and technical practices applied in a nuclear environment, and underline the importance of reliable industrial partnerships in supporting safe and stable long-term nuclear power plant operation.

Keywords: *nuclear power plant, main generator, auxiliary diesel generators, high-voltage motors, operation and maintenance, reliability*

IP-2

Automated Flaw Detection from Eddy Current Data Using Convolutional Neural Networks

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Eddy current testing (ECT) is a widely used non-destructive evaluation method for detecting surface and sub-surface flaws in conductive materials. Recent advancements in artificial intelligence and machine learning, particularly convolutional neural networks (CNNs) have shown great potential in automated flaw detection tasks. We present a novel CNN for the detection of anomalous eddy current inspection signals. To the best of our knowledge, the proposed method is the first such complex solution able to successfully analyze a real-world eddy current dataset and the first artificial intelligence automated analysis successfully qualified at EPRI. Our contributions are not limited only to the architecture of the CNN, but the overall construction of the experiment and the format of the input data. For CNN, we propose a 1D U-Net architecture for defect detection in multi-channel eddy current signals, leveraging a series of downsampling and upsampling layers with skip connections to capture both local and global features. The model was trained using the Focal Loss function to address class imbalance, with optimization performed via the Adam algorithm and a learning rate scheduler for gradual decay. To train the neural network, we utilized tube signals, indication positions, and indication widths as input features. Each neural network employs a specific filter through which the tube signal is processed. During runtime, preprocessing involves passing the input signal through a designated filter and segmenting it into chunks to prepare it for further analysis. We also employed ensemble learning, a methodology that integrates multiple neural networks to enhance predictive accuracy. Each network within the ensemble is trained on distinct datasets, optimizing its performance for specific types of inputs or indications. The outputs of these convolutional neural networks (CNNs) are subsequently aggregated to generate a unified and precise prediction. Neural network approach to automated analysis has half the amount of overcalls per tube compared to automated analysis approach with signal curve comparison INETEC has previously qualified at EPRI.

Keywords: *flaw detection, eddy current, automatic analysis, convolutional neural network, non-destructive testing*

IP-3

Opportunities for Nuclear Power in Maritime Applications

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Nuclear-powered propulsion for civilian applications has been demonstrated with several cargo ships (NS Savannah, Otto Hahn, Mutsu). However, there was no subsequent commercial deployment, partly due to the lack of public acceptance, and partly due to the lack of clear economic advantage. Today, a large percentage of global freight transportation is performed by large container ships, bulk carriers and oil tankers, contributing up to 3% of human-generated CO2. With the growing climate change concerns, it is increasingly clear that nuclear power has to be part of the solution. Moreover, maritime nuclear power may be deployed as Floating Nuclear Power Plants (FNPPs) with their specific advantages (and challenges). The presentation will discuss innovative approaches to nuclear-powered commercial civilian maritime applications aiming to enhance their competitive attractiveness.

Keywords: *maritime applications, nuclear-powered propulsion, floating nuclear power plants*

IP-4

The Role of Nuclear Power in Korea's Energy Mix: Current Status, Policy Direction, and Future Outlook

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This invited lecture provides an overview of the current status of nuclear energy in Korea and its role within the national energy mix. It introduces the structure and performance of Korea's operating nuclear fleet, recent developments in plant operation and lifetime extension, and ongoing efforts to maintain high standards of safety, reliability, and public trust.

Particular attention is given to the policy framework guiding Korea's energy mix, including the balance between nuclear power, renewable energy sources, and other low-carbon technologies.

Korea, a country with limited natiral resources, launched its national nuclear power program in the late 1950s. Beginning with the commercial operation of Kori Unit 1 in 1978, Korea is currently operating a fleet of 26 nuclear powe plants, which provide approximately 30% of the nation's electricity. Through joint design and technology cooperation with the United States, Korea achieved thechnology self-reliance and supply-chain and successfully exported nuclear power plants to the United Arab Emirates in 2009 and the Czech Republic in 2025. Looking ahead, Korea seeks to futher develop its nuclear industry as a strategic export sector and as a key pillar supporting sustainable national economic growth.

Korea has developed one of the world's most comprehensive and mature nuclear energy programs, with nuclear power playing a central role in ensuring reliable, low-carbon electricity supply. As of today, nuclear energy constitutes a significant share of Korea's electricity generation, supporting energy security, price stability, and climate change mitigation.

To ensure the stability of electricity supply and demand, Korea establishes a long-term (15-year) Basic Plan for Electricity Supply and Demand every two years. This plan forecasts future electricity demand and outlines the corresponding generation capacity and energy mix.

Under the 11th Basic Plan (2024–2038), two large-scale nuclear power plants and one small modular reactor (SMR) have been approved for new construction. In 2026, the government plans to formulate the 12th Basic Plan (2026–2040), which will serve as a mid- to long-term roadmap for managing electricity supply and demand in a stable and sustainable manner, with a strong emphasis on expanding renewable energy and achieving carbon neutrality.

The plan will more accurately project overall electricity demand, taking into account additional drivers such as the 2035 Nationally Determined Contribution (NDC) for greenhouse gas reduction, the growth of artificial intelligence (AI), data centres, and electrification. Based on these projections, it will establish

a carbon-free–oriented energy mix that carefully balances carbon neutrality, energy security, and economic efficiency. In this context, emerging technologies such as Small Modular Reactors (SMRs) and advanced reactor concepts are briefly introduced as potential contributors to future energy systems, especially in synergy with renewable energy sources.

Finally, the lecture shares key lessons learned from Korea's nuclear experience, including institutional arrangements, human resource development, and industry–academia–government cooperation.

Keywords: *Korea, Nuclear Role, Energy Mix, Policy, National Plan*

IP-5

Approach to License New Build NPPS in Czechia

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The effects of major changes on the energy market consolidate the market position of nuclear energy in Czechia and prove its importance for future in the energy mix. The lessons learned by the industry from the recent new build projects in the world will certainly be taken into account during the current nuclear build up and hopefully will help to avoid delays and cost overruns, which is often cited as a main drawback of the current nuclear industry development.

The country has prepared itself for the nuclear new build project for the last decade. One tender for two nuclear units in Temelin was already terminated due to the lack of financial guarantees to the project. State support for the Dukovany new build project, detailed preparation for the project and general preparedness of the industry and engineering capacities create a solid basis for the successful project implementation.

ENERGOPROJEKT PRAHA will play crucial role in licensing of the new NPP for its customer EDU II (Owner). Its responsibility is to prepare the complete set of documentation for licensing and permitting of the plant according to the atomic and civil construction laws respectively. Scheme of the process and cooperation of the Licensee and the EPC contractor in the framework of the regulatory procedures is described within the presentation.

Keywords: *licensing, newbuild, large reactors,energy, nuclear*

IP-6

Specific Issues of Licensing Small Modular Reactors

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Small Modular Reactors (SMRs) are considered as a promising option for enhancing energy security, supporting decarbonization goals, significantly increasing nuclear safety and enabling faster, economical and flexible deployment of nuclear power. Large expectations need to be confronted with realistic consideration of existing issues resulting from limited experience with deployment of SMRs. While many SMR concepts build upon proven reactor technologies, their innovative design features, deployment concepts, and operational modes introduce specific challenges for the existing nuclear licensing frameworks, which have historically been developed for large, site-specific nuclear power plants. The first part of the presentation will introduce general licensing issues associated with SMRs. These include the applicability of current regulatory requirements to SMR designs, the treatment of modularity and factory fabrication, multi-module sites, novel ownership and deployment models, siting flexibility, the potential use of graded or risk-informed approaches, etc. Particular attention is paid to the balance between maintaining established safety principles and adapting regulatory processes to accommodate innovation, while preserving the prime responsibility of the operating organization for nuclear safety. An overview of existing relevant regulatory efforts in selected countries and international frameworks will be provided.

The second part focuses on specific licensing challenges related to deterministic safety analysis (DSA) of SMRs. Key topics include the coupled analysis of neutronic, thermal-hydraulic, chemical, structural and radiological processes, identification and classification of postulated initiating events and accident scenarios reflecting innovative designs, validation of computer codes for unusual plant configurations, selection of design specific acceptance criteria, the definition of design basis and design extension conditions, the treatment of multiple reactors and shared systems, the effects of both internal and external hazards the specification of the levels and demonstration of compliance with the principles of defence-in-depth. The presentation will also discuss issues such as the selection of bounding scenarios, the role of passive safety features with specific failure modes, closer integration of deterministic and probabilistic approaches, the demonstration of practical elimination of early and large radioactive releases, and independent verification of safety analysis. The presentation aims to highlight areas where regulatory guidance may require clarification, adaptation, or further development to ensure a consistent, transparent, and robust licensing process for SMRs, while maintaining a high level of nuclear safety and public confidence.

Keywords: *Small modular reactors, nuclear safety, regulatory framework, licensing issues, deterministic safety analysis*

IP-7

CEZ Small Modular Reactors Program

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The presentation will provide an overview of the current status of the ČEZ project development, covering several aspects of the project: the need for SMRs in the Czech Republic, potential sites, design and supplier selection, project schedule, the ČEZ team, licensing and permitting, long-lead items, key design features of the Rolls-Royce SMR, and other relevant information related to ČEZ project development.

Keywords: *CEZ SMR project development*

IP-8

NUWARD SMR, a Multi-Energy Platform Serving Industries and Utilities to Accelerate Transition to a Low-Carbon Energy Future

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NUWARD, EDF's subsidiary, is advancing the development of its Small Modular Reactor (SMR) technology to address Europe's growing demand for safe, reliable, competitive, and abundant low-carbon energy. Market interest continues to rise, driven by the dual imperative of achieving Net Zero and ensuring long-term energy security. This is reflected in a growing number of inquiries covering diverse applications such as data centres, energy-intensive industrial processes requiring cogeneration, and urban district heating networks. To meet evolving customer needs and guarantee competitiveness, NUWARD has oriented its development toward a multi-energy platform capable of delivering sure, low carbon and affordable energy. The NUWARD SMR's reactor, with its thermal output of 1150 MWth, is designed to produce both electricity and heat delivering up to 400 MWe and 290 MWth in cogeneration mode. It is therefore fully adapted to meet the needs of European stakeholders. Designed in Europe, for Europe and beyond, NUWARD aims to become the leading European SMR energy platform providing low-carbon energy. The project is strengthened by robust partnerships with major European nuclear and industrial players, including Framatome, Arabelle Solutions, Tractebel, Ansaldo Nucleare, and Ansaldo Energia, enhancing its expertise in construction, modularisation, and innovation.

NUWARD also leads and contributes to several European initiatives, such as the International NUWARD Advisory Board and the Joint Early Review, an innovative initiative consisting of a joint pre-assessment of the NUWARD SMR carried out by NUWARD and eight European regulators. The Joint Early Review aims to best anticipate the expectations of safety authorities in several key European countries for its development. NUWARD is also a Project Working Group of the European Industrial Alliance on SMRs. All these initiatives aim to accelerate the deployment of SMR solutions and support market readiness.

Built on EDF's 60 years of experience in the design, licensing, construction, and operation of nuclear technologies worldwide, NUWARD SMR relies on proven technologies and robust industrial knowhow. By the early 2030s, NUWARD seeks to deliver a safe, competitive, and deployable SMR solution for Europe and subsequently for international markets.

Keywords: *NUWARD, SMR, cogeneration, district heating, data centres*

IP-9

KHNP i-SMR : Advanced Passive Safety Design and Deployment Strategy for Emerging Nuclear Markets

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The i-SMR developed by Korea Hydro & Nuclear Power is an advanced Generation 3+reactor designed to enhance safety, economic efficiency and flexibility for emerging nuclear market.

The i-SMR incorporates fully passive safety system based on natural circulation cooling and gravity-driven safety mechanisms, minimizing the need for active operator intervention during emergency conditions. With the integrated Control Rod Drive Mechanism-CRDM- the risk of control rod ejection accident is inherently eliminated.

Beyond technological innovation, KHNP's i-SMR deployment strategy integrates engineering, procurement, construction (EPC), operation & maintenance (O&M), fuel cycle management, and human resource development. Based on the strong supply chain we've built through large nuclear projects, KHNP already has a reliable supply system that can be applied to the i-SMR as well.

This presentation will introduce the key technical features of the i-SMR, including passively safety architecture, multi-hydraulic design concepts. It will also discuss KHNP's international cooperation roadmap aimed at supporting safe and economically viable SMR deployment in Europe.

Keywords: *i-SMR*

IP-10

Holtec SMR-300: From First-of-a-Kind Deployment to Global Fleet Readiness

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Holtec's SMR-300 is an advanced Gen III+ small modular reactor incorporating passive safety features and based on proven PWR technology, commercially available fuel, and established licensing frameworks. The design emphasizes inherent safety, constructability, and economic performance to support scalable fleet deployment.

Technical Highlights:

- High power density core; dual-unit configuration rated at over 680 MWe net output.
- Fully passive safety systems enabling walk-away safety (no off-site power, no off-site water, no immediate operator action) and simplified plant architecture.
- Exclusion Area Boundary, Low Population Zone, and Emergency Planning Zone contained within the Protected Area, supporting flexible siting near industrial or urban demand centres.
- Hybrid and fully air-cooled condenser options enabling deployment in water-constrained regions.
- High secondary steam superheat suitable for non-electrical applications.

SMR-300 is progressing toward construction readiness after more than 10 years of development. Major Nuclear Island design activities—including NSSS, containment, and passive safety systems—have been completed. Part 1 of the Construction Permit Application (CPA) was submitted to the U.S. NRC in December 2025 after extensive pre-application engagement initiated in 2019. In parallel, the design is undergoing the UK Generic Design Assessment (GDA), having completed Step 1 and progressing through Step 2, providing cross-validation of the safety case and alignment with European standards. Continuous regulatory engagement supports licensing predictability and construction efficiency.

The program builds on the First-of-a-Kind SMR-300 project at Palisades (Michigan, USA), scheduled for commercial operation in 2031. Engineering, licensing, long-lead procurement, and site preparation are underway. Palisades will serve as a reference plant supporting licensing, modular construction, supply chain qualification, and cost optimization for global fleet deployment. Manufacturing readiness is backed by Holtec's integrated U.S. industrial base.

Delivery is executed through a world-class consortium including Hyundai E&C (EPC), Framatome (fuel, licensing support and first core), and Mitsubishi Electric (I&C) under a turnkey, fixed-price model.

The SMR-300 represents a construction-ready and licensable pathway for secure, clean, and scalable nuclear generation.

Keywords: *Small Modular Reactor, Passive Safety Systems, Flexibility, Low Risk Licensing Approach, Turnkey Delivery Model, Non-Electrical Applications*

IP-11

Advanced Breakeven Molten Salt Fast Reactor (BeMFR) for Sustainable Nuclear Energy

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Amid rather optimistic expectations about roles of nuclear energy in battling global warming and climate crisis, there are also strong concerns about the sustainability of nuclear energy in a long-term view. In this talk, an ultimate nuclear reactor is pursued and suggested to address the essential nuclear trilemma, i.e., safety, spent nuclear fuel, and limited U resource. A breakeven molten salt fast reactor (BeMFR) is devised to fully resolve the nuclear trilemma in a competitive and reliable way. The BeMFR is a fast reactor using a chloride salt fuel and its fuel cycle is closed with the proliferation-resistant pyro-processing. The initial core of BeMFR can be loaded with U-TRU fuel. The U-TRU is directly recovered by the pyro-processing of the SNF stock without any TRU separation. In a new-comer nuclear country, an HALEU fuel can also used in the initial core. The reactor is operated such that the excess reactivity should not succeed about 1 dollar via on-line fission-product removal and feeding of metallized SNF. And the BeMFR core slowly transitions to an equilibrium breakeven one through a transition period. Depending on the initial core fuel, the operational transition strategies are different since the reactor characteristics are quite different. The detailed operational strategies are discussed to achieve the equilibrium BeMFR core. The safety and operational characteristics of BeMFR are evaluated for the whole operational period. It is shown that the SNF can be very efficiently reused and BeMFR can enable a long-term sustainable nuclear energy without relying on natural uranium. All the physics simulations are performed using both the Serpent 2 and iMC Monte Carlo codes.

Keywords: *Breakeven Molten Salt Fast Reactor, pyro-processing, spent fuel, closed fuel cycle, chlorination*

Round Table (RT)

SMRs today and tomorrow

Round Table Discussion will be moderated by: DAVOR GRGIĆ

Invited speakers will attend round table and discuss the following subjects:

- **When are you expecting to see first operating western SMR and where?**
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- **What do you think about passive systems as a first line of defence in SMR nuclear safety?**
- **What do you think about SMR and AMR common future?**

Session 1
Nuclear Safety Analyses (NSA)

S1-121

Overview of i-SMR Non-LOCA Safety Analysis (SAR sections 15.3&4)

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The innovative Small Modular Reactor (i-SMR) is a next-generation reactor model developed in the Republic of Korea. It features an integral reactor design with a double steel vessel structure, incorporating advanced characteristics such as soluble boron-free operation and a fully passive safety system to enhance both safety and operational flexibility. This paper presents the transient safety analysis of the i-SMR, performed using the SPACE thermal-hydraulic code and the THALES subchannel code. The study specifically focuses on two key accident scenarios: the Complete Loss of Reactor Coolant Flow (CLOF) and the Single Control Element Assembly Withdrawal (SCEAW). These two scenarios represent the most limiting cases in terms of fuel integrity evaluation for SAR sections 15.3 and 15.4, respectively.

The integral design of SMRs inherently limits the available internal space. This constraint requires the elimination of flywheels on the Reactor Coolant Pumps (RCPs), resulting in a rapid flow coastdown during a loss of power event. Such characteristics present a significant challenge to ensuring fuel integrity. While this poses a significant challenge to fuel integrity, the application of a soluble boron-free core ensures a negative Moderator Temperature Coefficient (MTC) throughout the entire burnup cycle, thereby effectively maintaining fuel integrity.

The increased control rod worth, resulting from the adoption of a soluble boron-free core, poses a potential threat to fuel integrity during a Single Control Element Assembly Withdrawal event. However, the core is effectively protected by the Core Quadrant Power Deviation reactor trip, which is triggered by in-core detectors that immediately sense the resulting power deviation.

The simulation results demonstrate the plant behaviour under these transient conditions and confirm that all relevant safety acceptance criteria are satisfied with sufficient margins, providing insights into the safety characteristics of the i-SMR design.

Keywords: *i-SMR, transient analysis, SPACE, complete loss of flow, single CEA withdrawal*

S1-122

Safety Analysis of i-SMR under LOCA Conditions using SPACE Code

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i-SMR (innovative Small Modular Reactor), currently being developed in Korea, is a new type of reactor that integrates the core, pressurizer, and reactor coolant pumps within a single vessel. Due to this integral design, large-diameter piping is eliminated, thereby fundamentally excluding the possibility of a LBLOCA (Large Break Loss of Coolant Accident). To satisfy the top-tier requirements of i-SMR—specifically, preventing core uncovery during design basis accidents—passive safety systems (PAFS, PCCS, and PECCS) have been adopted, significantly enhancing safety.

Safety analyses of i-SMR under SBLOCA (Small Break Loss of Coolant Accident) conditions were performed using the SPACE code, a thermal-hydraulic system code. In the event of an accident, reactor coolant is discharged into the containment vessel. Subsequently, PAFS and PCCS actuate to cool and condense the hot steam. This condensed coolant collects at the bottom of the containment vessel and is naturally injected back into the reactor vessel through the PECCS, driven by the pressure difference (hydraulic head) between the reactor vessel and the containment vessel.

The results confirmed that, even under conservative assumptions, the core remains covered, and stable conditions are maintained for 72 hours without operator actions. Consequently, i-SMR meets not only the regulatory acceptance criteria for LOCA but also the top-tier requirements. This demonstrates that i-SMR ensures improved safety compared to conventional large-scale PWRs.

Keywords: *i-SMR, LOCA, DSA, SPACE code*

S1-150

Protection System Setpoints for Steam Line Break Accident Containment Response Analysis for NPP Krško

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The break of a main steam line results in an initial increase in steam flow which increases the energy removal from the Reactor Coolant System (RCS) and, therefore, causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin and, potentially, a return to power. The core is ultimately shut down by the boric acid injection delivered by the Safety Injection System (SIS). A major steam line rupture is classified as an ANS Condition IV event, limiting fault. The following functions provide the necessary protection against a steam line rupture: reactor trip, SIS actuation, main feedwater isolation and steam line isolation. The steam line break accident is usually analysed first to prove that core parameters stay within design limits and second to prove that containment parameters (pressure and temperature) are within design limits. A double ended rupture of a steam line is the most limiting cooldown transient and is analysed at zero power with no decay heat to check core limits. The steam line break inside the containment may result in significant releases of high-energy fluid to the containment atmosphere. The mass and energy release depends on a number of different parameters, e.g., nuclear kinetics characteristics, stored energy in both the RCS and the main steam system, main and auxiliary feedwater operations, safety systems operation and break characteristics. Therefore, for each postulated break size different plant operating conditions prior to the break (power levels) should be considered in order to find the most adverse case regarding containment pressure and temperature. Usually, from point of view of containment parameters MSLB is limiting for containment temperatures (both atmosphere and liner temperature) and LOCA is limiting for containment pressure. In the paper the influence of changes in steam line break protection signal parameters (lead/lag constants) for steam line isolation and safety injection on containment pressure and temperature was evaluated. The steam line break was analysed using coupled code R5G. Coupled code R5G is a result of direct explicit coupling of RELAP5/mod3.3 and EPRI's GOTHIC 7.2b(QA) code. Large Double Ended Rupture (LDER) and split break were analysed for different power levels (30%, 70% and 102%). For split break, the maximum break area was found where the steam line isolation does not occur on low steam line pressure signal. The containment thermal hydraulic behaviour and design margins were calculated using GOTHIC part of the coupled code.

Keywords: *steam line break, protection setpoints, SIS actuation, containment limits, RELAP5/MOD 3.3, GOTHIC*

S1-151

Lock-up Characteristics of the Hydraulic Snubber for Plat Piping Systems

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Nuclear power plant piping is required to accommodate thermal expansion during normal operation while maintaining structural integrity under dynamic loading conditions such as seismic events and external disturbances. Excessive restraint of thermal movement may induce additional stresses in piping, whereas insufficient restraint during abnormal conditions can lead to excessive displacement and potential structural damage. Hydraulic snubbers are therefore widely employed as passive mechanical devices that allow slow thermal displacement while restraining rapid dynamic motion of nuclear piping. The performance of a hydraulic snubber is governed by its internal hydraulic behaviour, particularly the locking mechanism that determines the transition from a flexible response to a rigid restraint. A key performance parameter is the lock-up velocity, which defines the velocity threshold at which the snubber begins to provide restraint. However, the internal flow characteristics and pressure distribution governing the lock-up velocity are not always well understood, often resulting in conservative design approaches and uncertainties in operational reliability. In this study, a numerical investigation was conducted to analyse the internal hydraulic behaviour of a hydraulic snubber using computational fluid dynamics (CFD). A detailed internal geometry was modelled, and steady-state flow analyses were performed under various piston velocities. The pressure distribution acting on the valve components was evaluated, and the resulting hydraulic forces were compared with the spring reaction force to identify the lock-up velocity based on force equilibrium. The results show that the lock-up velocity of a hydraulic snubber can be predicted using CFD based on internal pressure distribution and force balance. The proposed analysis framework provides an effective approach for evaluating snubber performance and supports the design and assessment of hydraulic snubbers applied to nuclear piping. This study contributes to enhancing operational safety, long-term operation, and emergency preparedness of nuclear power plants by improving the understanding of passive vibration restraint devices.

Keywords: *Hydraulic snubber, Lockup-velocity, computational fluid dynamics (CFD)*

S1-153

Simulated Small-break Loss-of-Coolant Accident (SBLOCA): A Transient Analysis in PWRs

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Reliability analysis which takes into account multiple possible states are considered Multi-State reliability analysis. Reliability models for Multi-State systems allow different levels of performance in the system. Although multi-state reliability models provide more realistic and more precise representations of Nuclear Power Plant systems, their analysis is considered much more complex and more demanding. A popular approach to such an analysis consists in the application of stochastic processes, especially Markov chains. Unfortunately, the number of transitional state spaces in Markov chains grows exponentially with the increase in the number of component states. Algebraic Decision Diagrams prove to be an efficient data structure for storing Markov chains and can be used to perform such analysis. Preliminary results on smaller models show the applicability of algebraic decision diagrams to represent Markov chains, describing hundreds or even thousands of states from Nuclear Power Plant systems with multi-state components. These results justify the implementation of algebraic decision diagrams techniques to automate Markov chain analysis of systems with extremely large state space models. In order to substantiate the given conclusions, experimental results have been provided from a simplified prototype model of an algebraic decision diagrams based analysis of nuclear component cooling water.

Keywords: *Small Break Loss-of-Coolant Accident; Pressurized Water Reactor; Transient Analysis; Emergency Core Cooling System; Nuclear Power Plant*

S1-154

From Steam to Power: Using the GPWR Simulator to Strengthen Safety Culture in Turbine-Generator Operations

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Turbine-generator systems are a critical component of nuclear power plants, serving as the primary interface between thermal energy produced in the reactor and electrical power generation, while also supporting overall plant reliability and safety. These systems transfer thermal energy from the secondary side to mechanical shaft power, which is subsequently converted into electrical output. Effective turbine-generator operation requires operator situational awareness, accurate performance monitoring, and timely response to abnormalities to maintain stable plant conditions and prevent the escalation of transients that could challenge reactor safety. This paper examines how the Generic Pressurized Water Reactor (GPWR) simulator supports the strengthening of safety culture and knowledge management in turbine-generator operations through the simulation of realistic failure scenarios and the use of data-driven insights.

The GPWR simulator provides a controlled setting and safe environment where operators can practice handling turbine-generator failures such as loss of steam supply, pressure imbalances, and turbine trips. These scenarios provide high-pressure situations, giving the operators a safe way to practice handling the system faults without having the risks of real plant operations. The simulator allows operators to recognize alarm patterns, identify affected systems, and determine which components must be activated or controlled in accordance with established procedures, thereby strengthening system understanding and decision-making under abnormal conditions.

By collecting detailed data on operator actions and system responses, the simulator establishes a structured basis for knowledge development and performance assessment. These data support post-scenario analysis through event sequence review, parameter trend evaluation, and verification of procedural compliance, enabling plant personnel to examine operational responses, share insights, and strengthen safety protocols. Through systematic analysis of simulator outputs, operating procedures are refined, system reliability is enhanced, and continuous learning is integrated into routine practice. Operator performance is assessed based on adherence to established procedures, appropriateness of control actions, and effectiveness in maintaining stable plant conditions during abnormal and emergency scenarios.

The GPWR simulator incorporates detailed modelling of key secondary-side and balance-of-plant systems, including steam generator behaviour, turbine and condenser systems, feedwater control, steam admission and extraction processes, and turbine protection and trip logic. Integrating simulation results from these systems into training programs and operational reviews enhances knowledge transfer

across the organization and reinforces consistent operational practices. Overall, the study demonstrates how simulator-based training supports the systematic integration of operational experience into turbine-generator operations, contributing to sustained improvements in safety culture, knowledge management, and long-term plant safety and reliability.

Keywords: *Pressurized Water Reactor, safety, feedback, turbine-generator, decision-making*

S1-155

Calculation of Radiological Consequences of Design Basis Accidents for Licensing Purposes

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The calculation of radiological consequences of Design Basis Accidents (DBA) is used for licensing purposes including siting and SAR Chapter 15 applications. In this paper NPP Krško LOCA was used as a reference DBA accident. The released radiological effluents were calculated using Regulatory Guide RG 1.183 assumptions and RADTRAD computer code. Plant specific fuel source term is obtained using ORIGEN code for real plant's operation history data. The dispersion X/Q factors needed to perform calculation of doses in the environment (Exclusion Area Boundary EAB and Low Population Zone LPZ locations, and 1D/2D spatial dependence around the plant) were determined using two different approaches. The first is a classical one based on RG 1.145 as implemented in the PAVAN code with close range additions from RG 1.249 as implemented in ARCON96 code. The second is a X/Q calculation based on the Lagrange particle methodology developed by MEIS d.o.o. Both approaches use similar sets of meteorological data measured at NPP Krško location. Additional independent radiological consequences calculation was performed using MACCS2 code for the same release source term and meteorological data. The last methodology proposed to calculate TEDE doses around the plant was based on number of JRODOS calculations for selected past sequences of meteorological data, and statistical post processing of obtained spatial dose rate data. Three different approaches of radiological consequences calculation are compared from the point of view of accuracy, type of provided results, calculation time and user requirements. The methodologies were applied to existing Gen II NPP, but are mentioned to be applied to SMR plants with more focus on the radiological impact close to the plant too.

Keywords: *radiological consequences, DBA accident, LOCA, atmospheric dispersion*

S1-156

Strengthening Emergency Preparedness for Station Blackout Using a GPWR Simulator

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Station blackout (SBO) is a rare but potentially critical scenario that places significant pressure on nuclear power plant operators due to the loss of offsite electrical power, during which emergency diesel generators (EDGs) are automatically activated to supply essential safety systems. Despite EDG availability, SBO conditions still pose substantial challenges in maintaining key safety functions, requiring rapid decision-making, seamless coordination, and sustained operational control under uncertainty. Because such scenarios cannot be practiced during routine plant operations, effective emergency preparedness relies on realistic simulations and carefully structured drills. In this regard, simulator-based exercises are indispensable for developing operator competencies to manage SBO situations effectively.

This study aims to assess emergency preparedness and operator response during station blackout (SBO) events in a pressurized water reactor. To achieve this objective, a Generic Pressurized Water Reactor (GPWR) simulator was utilized to conduct scenario-based simulations that emphasize the practical challenges of emergency response under blackout conditions rather than theoretical analysis alone. The simulated scenarios begin with a loss of offsite power and progressively escalate to the failure of onsite alternating current sources due to emergency diesel generator malfunctions. Each scenario is designed to replicate realistic plant transients and demonstrates the behaviour of key safety systems such as the reactor protection system, engineered safety features, and emergency core cooling systems, as well as sequencer-driven actions and the increasing operational demands placed on operators as the event unfolds. The evaluation is based on qualitative observations and basic timing considerations commonly applied in emergency preparedness assessments.

The results indicate that effective preparedness for station blackout (SBO) events relies on early recognition of plant conditions and timely, conservative decision-making in accordance with established procedures. The simulated scenarios highlight the importance of structured response strategies and coordinated system management to maintain core cooling and ensure decay heat removal during prolonged power loss. Lessons from past nuclear accidents involving SBO conditions, particularly the Fukushima Daiichi accident, underscore how delays in response and limited preparedness can worsen accident progression.

This study demonstrates that the use of a Generic Pressurized Water Reactor (GPWR) simulator provides practical insights into emergency preparedness for SBO events. By modelling severe accident scenarios, simulator-based training enhances operator readiness by improving understanding of plant response be-

haviour, reinforcing adherence to emergency procedures, and clarifying response priorities under extreme power loss. These findings highlight the effectiveness of scenario-based simulation as a reliable tool for strengthening emergency preparedness programs in pressurized water reactor facilities.

Keywords: *station blackout, simulator, pressurized water reactor, emergency, safety*

S1-159

FDS Fire Modelling Calculations in NPP Safety Related Pump Rooms

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In case of fire protection engineering the results of the experiments or detailed fire protection calculations can provide us with fire related parameters (heat release rates, temperature distribution, heat flux and smoke concentration), fire propagation pathways and fire spreading rates which can be used to evaluate potential for damage/unavailability of the equipment designated as targets in fire compartment. In this paper FDS (Fire Dynamics Simulator) CFD code was used to perform calculation for two similar Krsko NPP safety related pump rooms having simple geometry and rather low thermal loading. The verified and validated fire modelling methodology utilizing performance-based approach with calculations required input data and potential benefits of performing such calculations for different fire scenarios were demonstrated. The one of selected rooms is the room with centrifugal charging (CS) pump belonging to CVCS system. The other one is room with safety injection (SI) pump belonging to SI system. Both rooms, in first approximation, have rather limited communications with other areas what makes calculation easier (fire rated door and ducts without fire damper). The difference between those two rooms is that SI pump room has safety related HVAC system located inside the room, and HVAC system for CS pump room is located outside the room. Combustible inventory in both rooms is lube oil, electrical equipment (motor parts, el. panels) and cable insulation (power, instrumentation and control). The pump lubricating oil is assumed as an ignition source in analysed fire scenarios. In one scenario, the heat input is modelled using prescribed time dependent heat release rate (HRR), and in other scenario using lube oil chemical composition to estimate burning rates. The room door is normally closed (lower opening is modelled), but it is modelled opened in one scenario 10 min after ignition by the fire brigade. There is no automatic fire suppression system in the rooms. The goal of the calculations is to model fire development and related spatial influence to be able to assess the possible fire damage of electrical cables in modelled fire compartment (target objects damage using temperature damage criteria) and to calculate the temperature at fire detector position. The smoke distribution and soot density are calculated too. This kind of fire modelling and obtained results provides more flexibility in achieving established performance criteria (so called performance-based approach) during all phases of nuclear power plant operations and it is related to achieving required level of nuclear safety.

Keywords: *performance-based fire modelling, FDS, NPP safety related pump rooms, heat release rate, fire compartment*

S1-161

Thermomechanical and Reactivity Feedback Analysis in Lead-cooled Fast Reactors

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Lead-cooled fast reactors (LFRs), identified as a key Generation IV nuclear technology, offer compelling advantages for sustainable energy production due to their inherent safety features, including operation at atmospheric pressure, passive heat removal capability, chemical stability of the coolant, and efficient fuel utilization. However, the severe LFR operating environment—characterized by strong temperature gradients, high irradiation exposure, and long residence times—raises important concerns regarding the mechanical stability of fuel assemblies and core internals. Thermally induced bowing and inelastic deformation may alter assembly-to-assembly contact conditions, impacting local power distributions, core restraint effectiveness, and ultimately reactivity and safety margins over the reactor lifecycle.

This work investigates the thermomechanical response of fuel assemblies in representative pool-type LFR core configurations, focusing on both single-assembly behaviour and ring-averaged core regions typical of European ALFRED- and LFR-AS-200-class designs. A simplified, yet physically representative, core model is adopted to capture the dominant across-duct temperature gradients and restraint-induced deformation patterns, which are key parameters for assessing the validity of point-kinetics-based neutronic-thermal hydraulic coupled calculations (NK–TK) under accidental transient conditions.

The simulation chain combines the NUBOW-2D INEL code for inelastic assembly mechanics with ATHLET system thermal-hydraulics, initially coupled through point kinetics, with provisions for higher-fidelity neutronic integration in future developments. Results highlight the sensitivity of assembly deformation and contact forces to restraint system configurations and thermal boundary conditions. Benchmarking against selected IAEA reference cases indicates reasonable agreement in predicted deformation trends, supporting the applicability of the approach for preliminary safety-oriented assessments.

To support systematic parametric studies and future multi-physics extensions, a lightweight Python interface (pyNubow) is employed to streamline model setup, execution, and post-processing, enabling improved integration with subchannel thermal-hydraulic tools such as DASSH.

Keywords: *thermomechanics, fast reactors, LFRs, reactivity coefficients, bowing*

S1-176

A Safety Analysis Methodology for Non-Loss of Coolant Accidents (Non-LOCA) in an Innovative Small Modular Reactor (i-SMR)

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This study presents a safety analysis methodology for Non-Loss of Coolant Accidents (Non-LOCA) in the standard design phase of an innovative Small Modular Reactor (i-SMR). The i-SMR adopts an integral reactor design in which the major components of the reactor coolant system (RCS), such as the core, reactor coolant pumps, steam generators, and pressurizer, are housed within a single reactor pressure vessel (RPV). This configuration eliminates large external piping, thereby inherently precluding large-break LOCA (LBLOCA). The compact system is enclosed in a steel containment vessel, enhancing overall robustness and safety. A conservative deterministic approach is employed for the safety analysis. System thermal-hydraulic behaviour under transient conditions is evaluated using SPACE (Safety and Performance Analysis CodE), a domestically developed system code accepted (or approved) by regulatory authorities. SPACE employs a multi-field two-phase flow model and an object-oriented structure, enabling accurate simulation of complex thermal-hydraulic phenomena with high modularity and reliability. For fuel safety assessment, the subchannel analysis code THALES is coupled with SPACE to calculate the Departure from Nucleate Boiling Ratio (DNBR), using a conservative, regulatory-compliant methodology based on an approved thermal margin model. The analysed events include Anticipated Operational Occurrences (AOOs) and Postulated Accidents (PAs), classified in accordance with the U.S. Nuclear Regulatory Commission's Code of Federal Regulations (10 CFR), and adapted to reflect the i-SMR's design characteristics. Acceptance criteria are established in accordance with design requirements, considering system pressure and temperature limits, fuel integrity, and radiological consequences. Conservative assumptions are applied for initial conditions, safety system actuation settings, single failure criteria, and loss of offsite power (LOOP). Decay heat is modelled using a recognized industry-standard correlation. Due to the fully passive safety systems and fail-safe design, no operator actions are required for accident mitigation, with a sufficient grace period provided for non-safety-related interventions. The results demonstrate that the i-SMR maintains system stability and fuel integrity under all analysed Non-LOCA scenarios. Key thermal-hydraulic responses remain physically plausible, and all safety criteria, including radiological release limits, are satisfied. These findings confirm that the i-SMR design provides adequate safety margins under design basis events and validate the proposed safety analysis methodology as a reliable basis for design verification and future licensing processes.

Keywords: *Innovative Small Modular Reactor (i-SMR), Non-LOCA, Safety analysis, Integral design, Passive safety*

S1-183

Validation of Coupled RELAP5/PARCS Codes for Main Steam Line Break analysis: a TMI-1 Benchmark Study

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The reliable prediction of reactivity transients in Pressurized Water Reactors (PWRs) requires a consistent representation of the strong coupling between reactor core neutronics and plant thermal-hydraulics. The Main Steam Line Break (MSLB) accident constitutes a particularly demanding scenario including the rapid and asymmetric overcooling of the reactor core, combined with the conservative assumption of a stuck-out control rod during the reactor trip, induces pronounced space-time effects and significant power redistribution phenomena. Such conditions challenge best-estimate analysis tools and provide a meaningful framework for assessing the capability of coupled multi-physics codes. In this work, the coupled RELAP5–PARCS Simulation Model SM is validated against the OECD/NEA MSLB benchmark based on the Three Mile Island Unit 1 (TMI-1) Nuclear Power Plant (NPP). The benchmark considers a full-power, end-of-cycle core configuration and has been specifically designed to evaluate the performance of advanced system codes in the simulation of reactivity transients involving multi-dimensional neutronic feedback. A detailed RELAP5 (SM) of the reactor coolant system was developed and dynamically coupled with a three-dimensional (3D) PARCS (SM) employing the benchmark-provided NEMTAB cross-section library. The coupling enables time-dependent evolution of power distributions and feedback parameters, allowing a physically consistent treatment of Doppler, moderator density, and temperature reactivity effects during the transient. The calculated time evolution of core power, reactivity, thermal-hydraulic variables, and steam generator behaviour is compared with benchmark reference results. In parallel, standalone RELAP5 SM with point kinetics are used to assess the impact of spatial neutronic modelling on the predicted transient behaviour. The results show that the coupled RELAP5/PARCS SM successfully captures the characteristic features of the MSLB scenario and a good agreement with benchmark data for both global plant thermal-hydraulic and core neutronic parameters, while minor discrepancies in the early phase of the transient can be attributed to modelling assumptions and code-specific correlations. The study confirms that a fully coupled multi-dimensional RELAP5/PARCS simulation is essential for an accurate representation of MSLB-induced space-time effects and for best-estimate safety analyses of Pressurized Water Reactor (PWR) reactivity accidents.

Keywords: *PWRs, MSLB, RELAP5, PARCS, coupling, TMI-1*

S1-189

Assessment of Intra-Pellet Temperature Distribution Effects on Reactivity During RIA Conditions: A Deterministic Analysis Based on the HERA JEEP Benchmark

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Reliable safety assessments of Light Water Reactors (LWRs) under Reactivity Initiated Accident (RIA) conditions require an accurate representation of feedback mechanisms governing core neutronics. Among these, the Doppler effect—strongly dependent on the spatial distribution of fuel temperature—plays a fundamental role in limiting reactivity excursions. While stochastic Monte Carlo codes can naturally account for detailed temperature distributions, conventional deterministic neutronics codes typically approximate the fuel pellet as isothermal, employing a single representative temperature for cross-section generation. This simplification may inadequately describe localized resonance self-shielding phenomena. This work presents an enhanced deterministic neutronics framework modified to explicitly account for the radial temperature distribution within the fuel pellet. The implemented methodology integrates detailed intra-pellet temperature profiles into the generation of homogenized nuclear data, bridging the gap between high-fidelity stochastic modelling and the computational efficiency of deterministic solvers. This enables a more physically consistent treatment of Doppler broadening and neutron interaction probabilities across different fuel regions within a deterministic environment.

To assess the impact of this approach, temperature distributions are derived from the High Burnup Experiments for Reactivity Initiated Accident (HERA) Joint Experimental Program (JEEP) Modelling and Simulation (M&S) exercise. Several temperature profiles, each representative of distinct temporal stages of a RIA transient, are considered. Although the present analysis is restricted to steady-state calculations, these snapshots reproduce thermally representative conditions of accident scenarios and allow for a systematic comparison between modelling assumptions. A reactivity-focused comparison is performed between the conventional single-temperature approximation and the distributed-temperature treatment. The analysis quantifies the differences in reactivity insertion associated with the two approaches and evaluates the sensitivity of the results to the severity of the temperature gradient.

The findings indicate non-negligible discrepancies in predicted reactivity when spatial temperature effects are explicitly modelled, demonstrating that the isothermal assumption may lead to an underestimation or misrepresentation of Doppler feedback under RIA-relevant conditions. While transient coupling capabilities remain the subject of ongoing development, the present study establishes a clear methodological foundation for high-fidelity accident analysis. The results emphasize the importance of resolving intra-pellet temperature distributions in deterministic reactor simulations and contribute to improving the reliability of safety evaluations within the HERA experimental framework and, more broadly, in LWR accident modelling.

Keywords: *RIA, Doppler feedbacks, LW, HERA*

S1-213

Uncertainty Quantification of the IBLOCA LSTF Test No.1 Based on the SAPIUM Guideline

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This study was conducted under the OECD/NEA ATRIUM program to evaluate and improve the SAPIUM UQ guideline. Focusing on critical flow and post-CHF phenomena in an IBLOCA scenario, uncertainty quantification (UQ) was performed for LSTF Test No. 1 (IET). Following the SAPIUM Guideline, validation databases were established and classified for inverse uncertainty quantification (IUQ) and validation. Key input parameters for each phenomenon were identified, and their uncertainties were determined through IUQ and verified via validation analyses. The UQ for LSTF Test No. 1 was executed in three steps: 1) considering only the two selected phenomena; 2) adding uncertainties in initial/boundary conditions, geometry, and material properties; and 3) incorporating core radial power distribution and CCFL. Results showed that while including more contributors improved experimental data coverage, the second-step factors had a negligible impact on PCT. Instead, CCFL and power distribution were confirmed as primary contributors. This study provides a technical basis for developing future IBLOCA safety analysis methodologies based on the SAPIUM framework.

Keywords: *uncertainty quantification, IBLOCA, LSTF, ATRIUM, SAPIUM*

Session 2
Operation, Maintenance and Lifetime Expansion Experience (OMLEE)

S2-106

Data-Driven Regression Approaches for Early Anomaly Detection in Medium-Voltage Electric Motors

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Reliable operation of medium-voltage motors is one of key prerequisites for availability of different equipment in nuclear power plants. Even with robust motor design margins, gradual degradation due to bearing wear, insulation ageing, reduced cooling performance and mechanical imbalance is still observed in these machines. Early symptoms often appear only as small variations in temperature, vibration or electrical load, which systems and procedures with fixed alarm thresholds typically do not detect. Possibility of more advanced diagnostic approach was studied. For several critical motors, ten years of measurements are available, including winding and bearing temperatures, vibrations, currents and relevant system parameters. The reconstruction of normal operating behaviour and the identification of deviations linked to developing faults are possible using this data. Before modelling can be performed, the dataset is preprocessed by removing corrupted records, interpolating missing values and normalizing variables. To enhance early fault detection, regression-based anomaly-detection models are applied. The models are trained exclusively on periods representing healthy operation, learning the relationships between environmental, system and motor-specific variables. Once trained, models generate expected values for key parameters such as bearing temperature. Deviations between predicted and measured values serve as indicators of abnormal behaviour, particularly when residuals show persistent shifts, exceed statistical limits or trigger cumulative deviation metrics. A portion of the historical dataset is withheld for validation to assess model accuracy on previously unseen data. Metrics such as the coefficient of determination and root mean square error are used to quantify predictive performance, while fault detection capability is evaluated on periods containing simulated fault conditions. The results show that regression models can detect subtle anomalies significantly earlier than traditional threshold-based systems, providing maintenance teams with valuable time to plan inspections or adjust monitoring strategies. The integration of regression models for early warning detection is regarded as a promising enhancement to maintenance practices and is considered to support the long-term reliability of critical rotating components.

Keywords: *Medium-Voltage Induction Motor, Predictive Maintenance, Regression Model, Neural Network, PCA*

S2-107

Dielectric Losses Diagnostic Testing of Medium-Voltage Electric Cables and Motors in Krško NPP

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In this paper medium-voltage power cables supplying medium-voltage induction motors in a nuclear power plant were addressed. This article presents 10 years of testing and results on how diagnostic outcomes are influenced when cable circuits are tested stand-alone versus with the motor connected. The insulation systems (e.g., EPR and XLPE) and their typical ageing stressors were outlined, and the customary diagnostic toolbox (insulation resistance - IR/ polarisation index - PI, dielectric loss - TD/tan delta, partial discharge - PD, and complementary frequency-domain approaches) was summarised. Dielectric-loss testing was employed at very low frequency (0,1 Hz) and at power frequency (50 Hz), where very low frequency measurements were applied using a portable device, whereas 50 Hz arrangements were treated as more demanding in terms of test infrastructure. Measurement campaigns were described and discussed in which medium voltage cables were tested disconnected from the load and the same circuits were tested with the asynchronous motor connected. The tests were performed using stepped voltage tan delta procedures at $0{,}5 \cdot U_0$, $1{,}0 \cdot U_0$, $1{,}5 \cdot U_0$ and $2{,}0 \cdot U_0$; where feasible, the motor insulation was additionally assessed separately at 50 Hz. The results were compared quantitatively and by trending. It was shown that the connected motor could greatly influenced the loss response and increase measured tan delta by approximately one order of magnitude, which had hidden the cable contribution and required disconnection for cable-condition assessment. As an alternative to full disconnection, a guarded very low frequency configuration (Guarded In/Out, cable and motor) was considered to separate cable and motor contributions within a single test set-up. Finally, a decision-oriented evaluation approach was provided, and the advantages and limitations of the tested configurations were summarised with respect to sensitivity to moisture/ageing at very low frequency, representativeness at 50 Hz, mobilisation effort, and interpretability against acceptance criteria.

Keywords: *nuclear power plant, medium voltage, cable, electric motor, testing, dielectric loss-tan delta*

S2-114

Modernization of Central Alarming System at Krško NPP

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The existing central Alarm System, model SER4100 manufactured by Qualitrol, has been in operation since 2004, with upgrades implemented in 2008 and 2018. Increasingly frequent faults—such as system unresponsiveness, insufficient processing power, and loss of communication within the system components—have reduced the overall reliability of the AS system. Due to the lack of technical support and the limited availability of spare parts, replacement of the existing alarm system was required.

The upgrade was carried out during the 2025 outage and covered the entire system, from the input terminals—which remained unchanged and defined the project boundary—up to and including all alarm panels (ALBs) on the main control board (MCB). The system is integrated with a modern distributed control system (DCS), manufactured by RTP and designed with two redundant channels. The new system allows live replacement of all vital components ("hot-swap"), ensuring maximum system availability in the event of failures. Owing to the redundant two-channel design, if one channel fails, the other maintains full system functionality and enables uninterrupted plant operation.

The new alarm panels on the main control board are LED-lit and digitally connected to the main system via a redundant MODBUS protocol. This allows the use of network cabling, easy reconfigurations and significantly shortens installation time compared to the old hardwired, bulb-lit ALBs. The system was fully assembled and configured at the manufacturer's facility, enabling comprehensive FAT testing. Such approach reduced testing activities during the outage and increased reliability during system startup and operation.

This impressive modernization project, covering nearly 2,000 digital inputs and controlling 33 ALBs (1,260 alarm windows), was fully installed and tested in only eight days with no major issues.

Keywords: *Alarm System (AS), Distributed Control System (DCS), Redundant architecture, Alarm Light Box (ALB)*

S2-124

Overview of NEK Commercial Grade Dedication Process With Regard to Industry Lessons Learned

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Nuklearna Elektrarna Krško (NEK) operates a Pressurized Water Reactor (PWR) in which the integrity and long-term reliability of the secondary water–steam cycle represents a critical factor for safe and efficient plant operation. During extended refuelling outages, secondary-side components are exposed to atmospheric conditions, significantly increasing the risk of corrosion and deposit formation. To mitigate these effects, NEK implemented Film Forming Amine (FFA) technology for the first time in 2021. While the initial application demonstrated positive chemistry-related effects, operational challenges were observed, primarily associated with dosing skid limitations and feedwater flow measurement disturbances.

Based on lessons learned from the first campaign, a comprehensive modernization of the FFA dosing skid was performed prior to the second application in 2025. The upgrade focused on improving system reliability, operational controllability, and chemical dosing accuracy. Key modifications included redesign of the suction and flushing lines to enable effective rinsing after chemical injection, improvements to the mixing tank drainage concept, optimization of heating control through relocation and modification of temperature sensors in the dosing water lines, installation of alternative level measurement and visual inspection ports to overcome condensation-related radar sensor failures, and implementation of enhanced cleaning and service procedures following dosing completion. Factory Acceptance Tests (FAT) and Site Acceptance Tests (SAT) were conducted to verify functional performance and instrumentation response prior to site deployment.

The refreshed FFA application was executed between September 2 and September 18, 2025, comprising 13 dosing days with controlled pauses. A total of 750 Liters of 5 wt.% ODACON®F emulsion was injected into the secondary cycle while maintaining free FFA concentrations below the defined operational limit of 300 ppb. Real-time adjustment of dilution ratios and pump stroke settings enabled stable concentration profiles and prevented anomalies in feedwater flow measurements, which had been a major concern during the first application.

Operational and chemistry data confirmed that no adverse effects on plant performance occurred during the injection period. Increased steam generator blowdown flow supported removal of mobilized corrosion products, while pH, conductivity, ammonia, and hydrazine parameters remained within specification limits. Visual inspections performed during the subsequent refuelling outage revealed a pronounced water-beading effect on several secondary-side components, indicating successful formation of a protective hydrophobic film.

The results demonstrate that targeted skid modernization combined with an optimized dosing strategy significantly improved the robustness and operational compatibility of FFA application at NEK. The experience gained provides a validated technical framework for future FFA campaigns and supports broader implementation of film forming technologies in PWR secondary systems.

Keywords: *Film Forming Amine (FFA), FFA injection, Secondary Side Chemistry, PWR Water–Steam Cycle, Corrosion Mitigation, FFA Dosing Skid, Feedwater System, Condensate System, Outage Chemistry, ODACON®, Nuclear Power Plant Krško*

S2-136

Digital Rod Position Indication Advances Display System Installation at Krško NPP

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Pressurized Water Reactors utilize neutron-absorbing rods for rapid changes in reactivity. Control and shutdown rods are arranged into Rod Control Cluster Assemblies (RCCA), which are grouped into banks. Each RCCA is attached to its grooved drive rod, which is being inserted or withdrawn by the rod drive mechanism. It is of utmost importance to know the actual position of the RCCAs and the eventual misalignment. Per the original design, Krško NPP utilizes the Digital Rod Position Indication System (DRPI) to continuously monitor the position of all 33 control and shutdown RCCAs and present their position to the operators in the Main Control Room. The position of an RCCA is sensed by rod position detector coils mounted outside of the rod travel housing. Coils are arranged into two separate trains. Sensed position of each train is processed by its data cabinet and combined in the single control board display to achieve full accuracy. Being designed in the 1970's, the original DRPI display had some limitations due to the technology available at that time. Some of its failures required a plant shutdown per Technical Specifications Limiting Conditions for Operation. Based on positive industry experience and several reference installations, Krško NPP replaced the legacy DRPI display with a new product from the original system vendor. DRPI Advanced Display System (DADS) introduces redundancy in all potential single points of failure while maintaining full system accuracy. The new DRPI display design is based on two flat-panel displays and FPGA-based controllers mounted on the associated drawer assemblies, which display all RCCAs and alarms across both displays. Automatic fallback of all rods' display to the operating display in case of a single DRPI display or drawer failure is provided. DADS has also introduced monitored redundant hot-swappable power supplies with fault indication, and an additional redundant Ethernet interface, which provides alarms alongside rod positions to the plant computer. Plant Technical Specifications require rod drop time measurement every refueling outage to ensure that rods insert properly and within the time response requirements of the plant's safety analyses. DADS includes automatic integrated rod drop testing with report generation, so that no other equipment or personnel is needed. Maintenance-friendly design allows for replacement, maintenance, or online troubleshooting on one DADS drawer while the other drawer still provides full indication. Installation and commissioning were carried out during the Outage 2025.

Keywords: *rod position indication, DRPI, rod control, DADS*

S2-138

Water Treatment System Replacement in Perspective of Long Term Krško NPP Operation

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Quality of a demineralized water for plant needs deteriorated over time and some components were losing their performance. Adding additional mixed bed and replacement of selected components like reverse osmosis membranes and electro deionisation units were an option, however in perspective of possible operation even beyond 60 years of the plant contributed to a decision to replace practically the whole system. The system replacement took a longer period of time and therefore a temporary mobile water treatment plant was used to supply demineralised water to the plant during the demolition, installation and testing period. Project finally resulted in stable high quality demineralised water in automatic operating mode without constant presence of operators. All relevant chemistry parameters of demineralized water are kept extremely low and in accordance with the best industry standards and praxis. Chemistry parameters create chemistry performance indicators which is further input to calculate plant performance indicators as defined by WANO (World Association of Nuclear Operators). The system is controlled via an upgraded man machine interface from a local operating console with connection to a main plant control room. Compatibility with other plant system overview and control is assured. Stable high-water quality has a potential to significantly contribute to control of corrosion processes of vital plant components like steam generators and other plant systems, both on primary as well as on secondary site in general. this is of a great importance in perspective of long term NPP operation, especially beyond 60 years plant lifetime.

Keywords: *demineralised water, long term operation, corrosion, chemistry parameters, performance indicator*

S2-158

JEK2 Condenser Cooling Technology

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GEN energija d.o.o. is currently evaluating the construction of the second unit of the Krško Nuclear Power Plant (JEK2). The proposed site is situated west of the existing nuclear facility, nestled in the Sava River valley on the western edge of the Krško plain. The important part of the new nuclear power plant (NPP) will be its cooling system. Because the thermal capacity of nearby Sava River is already used in full extent for cooling of the existing power plant, the new NPP will have to rely on cooling towers. Approximately two-thirds of the heat produced in the reactor of the nuclear power plant is dispersed to the environment through the cooling system, only one-third is converted into useful mechanical work via turbine. Efficient heat dispersion from the condenser is vital to maintaining the necessary enthalpy potential for turbine operation. Given the long operational lifespan of a nuclear power plant, even marginal optimizations in cooling efficiency result in significant cumulative energy gains. Systems responsible for cooling power plants have a significant influence on spatial planning. Cooling towers occupy a considerable portion of the total plant layout, and some cooling towers can reach heights up to more than 200 meters. These systems also have an environmental impact due to their consumption and discharge of water to and from river. This is why this topic is important, even in the initiative phase and in national spatial planning. This paper evaluates and compares various cooling solutions suitable for the condenser cooling of JEK2. Four distinct technologies—natural draft, mechanical draft, fan-assisted natural draft, and hybrid cooling towers—are identified, analysed, and compared. The study emphasizes the importance of selecting a cooling technology that aligns with both the specific technical parameters of the plant and the surrounding environmental constraints.

Keywords: *Cooling, Cooling tower, JEK2, NPP*

S2-160

Evaluation of Control Rod Drop Time Measurement Methodologies in Nuclear Power Plants

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Nuclear power plants use neutron absorbers to control the magnitude of neutron flux in an operating nuclear reactor. Essential part of that control mechanism are Control and Shutdown Rods, which control the neutron flux by changing the length of the neutron absorber presented inside the reactor core. This is commonly referred to as Control Rod Height. Naturally, power plant manufacturers devised a system to indicate actual rod height to the control room operators, called Rod Position Indication system (RPI). Same system is used to measure the time required for Control and Shutdown rods to go from fully withdrawn to fully inserted position (Rod Drop), ensuring swift and proper shutdown of the nuclear reactor. As such, this Rod Drop Time measurement is required on a plant cycle basis to certify the plant for power operation. As the digital age progresses, innovation and introduction of digital systems has made alternative methods of Rod Drop Time measurement feasible. This paper evaluates the feasibility of the new measurement methodology and compares it with the legacy Rod Drop Time measurement, still widely used in nuclear power plants.

Keywords: *nuclear, rod drop, control rods, rod position indication, RPI*

Session 3
Nuclear Option in the Context of Energy, Economics, Finance and Resilience (NEEFR)

S3-101

Powering Small Grids and Islands Using Floating Nuclear Power Plants

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The emergence of small modular and micro reactors has brought new possibilities, in general, but perhaps even more so to small power grids and islands that have historically been considered too small to even contemplate using nuclear power. With power output ranging from 5 MWe to 300 MWe, small modular and micro reactors can be adopted to a very large range of usage. In this paper, the possibility of using floating nuclear power plants is being considered and analysed. Such floating nuclear power plants have many similarities with ships, but they deliver power to grids and not to propulsion. Furthermore, being anchored up in one jurisdiction makes the nuclear regulatory aspects much easier than for ships. However, they share the benefits of being build and decommissioned on a single shipyard as opposed to on-premises, and they resist earthquakes and tsunamis very effectively. However, being operated by a small electricity grid, or even islands, the manning situation is not very different from a ship. For a small grid or an island, recruiting enough people to operate a selfsustained nuclear power plant is difficult. Hence, by using the same types of reactors that are suitable to those on ships, small grids and islands should have a realistic option. In the NuProShip I project, such reactors are selected after subjecting them to 37 criteria. However, with a focus on ships, an analysis of a floater for electricity production is not performed. This paper will present such an analysis. The technologies in question are helium gas-cooled reactors, molten-salt reactors and lead-cooled reactors. However, only certain kinds are relevant due to proliferation and security concerns that also can be found in small grids and island. TRISO fuels has therefore been chosen for the helium gas-cooled- and the molten-salt reactors, and uranium nitrate for the lead-cooled reactor due to their proliferation resistance. The purpose of this paper is to perform a high-level techno-economic analysis of the proposed floater using different configurations of reactor technologies and number of reactors ranging from the very smallest of 5 MWe to a double installation of lead-cooled reactor giving in total 110 MWe or higher with more units. Levelized Cost of Energy will be used as basis of comparison, and an uncertainty analysis is provided.

Keywords: *High Temperature helium Gas-cooled Reactor (HTGR), Molten Salt Reactor (MSR), Levelized Cost of Energy (LCOE), Liquid lead Fast Reactor (LFR), TRISO.*

S3-102

Stakeholder Engagement in the Adoption of Nuclear Propulsion in Merchant Shipping

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The successful attainment of the EU's 2030 Climate Target Plan necessitates substantial reductions in carbon dioxide emissions from international maritime transport. Moreover, the International Maritime Organization has adopted a strategy aimed at achieving net-zero greenhouse gas emissions from international shipping by 2050. These objectives entail both a reduction in energy consumption through enhanced energy efficiency and a transition towards cleaner types of energy, including renewable energy and low-carbon fuels. Nuclear propulsion in merchant shipping represents a novel and potentially transformative technology with the capacity to substantially reduce greenhouse gas emissions from international maritime transport. It enables long operational intervals without refuelling and can support either higher transit speeds or increased cargo capacity, depending on the priorities of the shipping companies. At the same time, the technology introduces a distinct set of challenges, which are examined in this study through a comprehensive PESTLE analysis. Although the technical and regulatory challenges are substantial, the societal dimension plays an especially critical role. As highlighted by the International Atomic Energy Agency, stakeholder engagement is a vital element of nuclear projects. Unlike land-based nuclear facilities, where stakeholders are typically fixed, nuclear propulsion in merchant shipping involves a dynamic stakeholder landscape, as vessels move between ports and jurisdictions. These stakeholders must therefore be identified and engaged. Understanding stakeholder perspectives, particularly those of crew members, is essential for the responsible and successful implementation of nuclear-powered merchant shipping. Although nuclear propulsion in merchant shipping is a novel application, apart from a few merchant ships in the past, concerns stemming from past nuclear incidents remain and must be addressed. This study identifies the key stakeholders in nuclear-powered merchant shipping, with a particular focus on the maritime workforce. It examines the level of acceptance of nuclear-powered vessels, as well as the safeguards and operational conditions they deem necessary for considering employment aboard such ships. The significance of this research lies in its contribution to an emerging and underexplored area of maritime innovation – nuclear propulsion in merchant shipping – by identifying the prerequisites for successful implementation and addressing a critical human factor: crew acceptance.

Keywords: *Nuclear propulsion in commercial shipping, stakeholder engagement, nuclear corridor, crew acceptance, safeguards.*

S3-123

System Levelized Cost of Electricity of Nuclear Power Plants and Other Energy Sources

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Within energy system analysis, there is discourse regarding the role and economic benefits of nuclear energy in terms of overall system costs. The reported findings range from considerable drawbacks to substantial benefits, depending on the chosen models, scenarios, and underlying assumptions.

Levelized Cost of Electricity (LCOE) is a common metric of comparing various power generating technologies. However, the mass introduction of variable renewable energy sources, including wind and solar photovoltaic, leads to additional costs caused by intermittency, the so-called integration costs. This paper will try to investigate and re-define the concept of system LCOE and proposes concrete methods to estimate it for each technology.

This system LCOE will try to allow for the economic comparison of various generating technologies and derive appropriate metrics. Integrating low-carbon nuclear-based power generation technologies to balance the fluctuating supply from renewable sources is becoming increasingly important in the future energy mix, as well as that integration costs increase with growing renewables shares and become an economic barrier to deploying renewable energy sources at high shares.

The goal of system LCOE is to help understand and resolve the challenge of integrating hybrid energy systems and can guide research and policy makers in realizing a cost-efficient transformation towards an energy system with a mix of nuclear power and a high share of variable renewables.

Keywords: *LCOE, integration costs, hybrid energy systems, System LCOE, energy mix*

S3-126

JEK2 Project Schedule: Key Milestones Toward the Final Investment Decision

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The JEK2 project is a key strategic investment to strengthen Slovenia's long-term energy security, reduce carbon dioxide emissions, and support the transition to a low-carbon energy system. This paper presents the project timeline up to the Final Investment Decision (FID), planned for 2028, including spatial planning, safety documentation, supplier selection, and financing arrangements.

The project timeline spans from 2025 to around 2040, with the start of operation of the new nuclear unit planned for around 2040. A major milestone in 2026 is the decision, expected to be adopted in the first quarter of the year, to initiate the official preparation of the National Spatial Plan (NSP) for JEK2. Key future milestones include the FID in 2028, contract signing with the selected supplier in 2029, issuance of the Integrated Construction Permit and first concrete pouring in 2032, and the start of operation around 2040.

The decision to initiate preparation of the National Spatial Plan is expected to be adopted in the first quarter of 2026. Spatial planning formal procedure started on 1st of July 2025 through publication in the Ministry of Natural Resources and Spatial Planning Spatial Information System and is carried out in accordance with the Spatial Planning Act (ZuREP-3). The public consultation period lasted until 31 October 2025 and included a series of project presentations at both, national and local, levels. Following the public consultation, the preparation of the Guideline Analysis began, and the decision to initiate the preparation of the National Spatial Plan for JEK2 is expected at the end of January or in early February 2026.

Safety analysis documentation preparation started in 2024 with the Site Safety Analysis Report (SSAR), followed by radiological impact assessments, Probabilistic Seismic Hazard Analysis (PSHA), and Probable Maximum Flood analysis (PMF). These activities are expected to be completed by the end of 2026 or in the first half of 2027 and will support applications for construction and operating permits.

Supplier selection is based on NOAK technology with a single reactor unit. The Technical Feasibility Study (TFS) was completed in 2025, while supplier qualification, tendering, bid evaluation, and negotiations are planned to conclude by the end of 2028, in line with the FID. Contract signing is planned for 2029.

The preparation of the financing model builds on completed economic and technical analyses. Within the Working Group for JEK2 on the level of Government of Slovenia, a Report on the Preparation of the JEK2 Project Financing Model was prepared and approved in November 2025 by a core working group led by a representative of the Ministry of Finance and is currently awaiting official publication. To maintain the

project schedule, early agreement on further development of the financing model will be required, together with the preparation of documentation for notification at the European Union level, a process that is demanding and may take two years or longer.

The project schedule is considered feasible, with potential for acceleration through legislative adjustments or the adoption of a special law. Successful implementation will require strong stakeholder coordination, timely decision-making, and effective risk management.

Keywords: *JEK2, Project Schedule, Final Investment Decision (FID), Nuclear Energy*

S3-127

Non-electric Applications of Small Modular Reactors in Slovenia

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Small Modular Reactors (SMRs) are increasingly recognised not only as low-carbon electricity sources but as flexible, multi-purpose energy systems capable of delivering heat, hydrogen and selected industrial energy services. Their ability to operate in cogeneration mode and to supply thermal energy across a broad temperature range enables applications that extend beyond electricity generation.

This paper provides a concise overview of non-electric applications of SMR technology, with particular emphasis on those identified as most relevant for the Slovenian energy system. The analysis indicates that SMRs can represent a technically robust platform for the decarbonisation of energy-intensive sectors and for the integration of low-carbon heat into existing energy infrastructures.

A key challenge of modern decarbonisation strategies is that, while electricity generation can increasingly rely on renewable and nuclear sources, a large share of final energy demand remains in the form of heat, steam and chemical energy carriers. In Slovenia, these demands are primarily associated with district heating systems and industrial processes, which continue to depend largely on fossil fuels. SMRs offer a pathway to address this gap by supplying both electricity and significant amounts of usable thermal energy from a single low-carbon source.

The enabling mechanism for non-electric applications is nuclear cogeneration. By simultaneously producing electricity and useful heat, SMRs can achieve higher overall energy utilisation compared to electricity-only plants. Partial extraction of reactor thermal power allows substitution of fossil-fuel boilers and furnaces while maintaining stable electricity generation, contributing to emissions reduction and improved system efficiency.

District heating is identified as one of the most mature and near-term non-electric applications. SMRs can supply hot water or low-pressure steam in the temperature range of approximately 80–150 °C required by district heating networks, providing a reliable and weather-independent heat source. In the Slovenian context, heat demand is concentrated in urban areas with existing or planned district heating infrastructure, making this application particularly attractive for early implementation.

Industrial process heat represents another high-impact application. A significant share of industrial energy consumption is linked to the generation of steam and process heat for chemical reactions, drying, distillation and material processing. SMRs can deliver thermal energy from low-temperature hot water up to high-temperature steam exceeding 500 °C in advanced designs. Analyses indicate that industries with continuous heat and steam demand—such as chemical processing, pulp and paper, food and beverage production, pharmaceuticals and selected metallurgical processes—are among the most promising potential users in Slovenia.

Hydrogen production is also strategically relevant. SMRs can support hydrogen generation through low-temperature electrolysis using nuclear electricity as well as through high-temperature electrolysis or thermochemical cycles using nuclear heat, with the latter offering higher efficiencies. In Slovenia, hydrogen is primarily considered for industrial consumption rather than large-scale export. SMRs can additionally supply thermal and electrical energy for desalination, although this application is of limited relevance under current national conditions.

Overall, the analysis confirms that the primary value of SMRs in Slovenia lies in their ability to provide reliable, low-carbon heat and cogeneration for urban and industrial clusters, complementing electricity generation and enabling deeper decarbonisation of the national energy system.

Keywords: *SMR, Licensing, sitting, financial analysis*

S3-134

Microreactors: Enabling Resilient Energy Solutions for Critical Applications

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Microreactors are emerging as a transformative option within the Small Modular Reactor (SMR) portfolio, offering compact, transportable, and factory-fabricated solutions to deliver reliable, low-carbon energy where conventional infrastructure is impractical. Their unique attributes, such as autonomous operation, passive safety, and extended refuelling intervals, position them as a strategic enabler for resilient energy systems in remote communities, industrial sites, and specialised applications such as space missions. By replacing diesel generators and allowing companies to meet climate targets, microreactors offer a practical pathway to decarbonise niche sectors such as data centres and mining operations. This paper examines the technical and strategic foundations of microreactor development, highlighting design features that enhance simplicity, safety and flexibility, including passive heat removal and modular deployment concepts that enable rapid installation with minimal site preparation. These characteristics support integration with microgrids, complement renewable energy sources, and provide highly reliable baseload power for critical applications. Beyond electricity generation, microreactors can enable non-electric uses contributing to sustainable development goals. Central to this discussion is the role of the International Atomic Energy Agency (IAEA) in supporting Member States to harness microreactor technologies for their specific objectives. It highlights IAEA initiatives in technology mapping and information sharing through tools such as the Advanced Reactors Information System (ARIS) and the SMR Booklet, which consolidate global design data and deployment trends. In addition, the Agency is preparing a dedicated TECDOC on microreactor technology developments, scheduled for publication in 2026, to provide Member States with a comprehensive reference on design features, deployment models, and cross-cutting issues. These efforts aim to provide technical guidance, harmonise approaches, and facilitate knowledge exchange to accelerate safe and secure deployment. The Agency also promotes international collaboration through technical meetings, Coordinated Research Projects, and capacity-building programmes. Together, these initiatives create a foundation for informed decision-making and foster confidence in adopting microreactor technologies for specialised applications. By integrating technical insights and collaborative frameworks, this paper demonstrates how microreactors can become a cornerstone of resilient energy strategies for specialised applications. Their successful implementation will depend on coordinated action across technology development, regulatory adaptation, and international partnerships, areas where the IAEA plays a pivotal role in shaping the future of nuclear energy.

Keywords: *Microreactors, Reactor Technology, SMRs, Resilient Energy Systems, International Collaboration*

S3-144

Maritime Applications of Nuclear Energy

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We live in a time of great technological change and numerous challenges. These changes affect all aspects of our lives. Unfortunately, their impact on the environment and the resulting pollution is inevitable. Of particular note is atmospheric pollution caused by greenhouse gas emissions. Despite the efforts of the International Maritime Organization to reduce greenhouse gas emissions, in 2024 global maritime transport emissions amounted to 973 million tons of CO_2, representing an increase of 9.4% compared to 2019.

Maritime transport plays a leading role in global transport, carrying more than 80% of goods by volume. Like most technologies, shipping is already experiencing, and will continue to undergo, major changes in the future. It is assumed that seafarers will require skills in the future that are not currently necessary. In order to decarbonize maritime transport, a greater increase in the use of nuclear energy in shipping can be expected in the future. As it is expected that the occupational profile of a seafarer will change in the future and require new and different knowledge, this may include knowledge related to the use of nuclear energy to power ships.

The use of nuclear energy brings with it some additional aspects that require additional attention. These are: high initial costs, increased safety requirements, the crew must have some additional knowledge, the need for additional crew training, the problem of waste storage, and the social acceptability of nuclear energy. Despite these aspects the use of nuclear energy offers great advantages such as a secure energy supply, low fuel costs, the ability to deliver large amounts of energy, minimal CO_2 emissions and savings in ship space. For all of the above, additional attention should be paid to the development of the use of nuclear energy in maritime transport. Additionally, from a maritime transport perspective, special attention should be paid to modular nuclear reactors.

Keywords: *maritime transport, zero-emission, nuclear energy, CO_2 emissions*

S3-201

Methodology for the Identification of Sites for the Deployment of Small Modular Reactors (SMRs) in Slovenia

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The Republic of Slovenia is strategically committed to the decarbonization of its energy sector and to complementing existing and planned nuclear capacities with advanced modular technologies. The objective of the study is to develop an expertly justified and evidence-based set of potentially suitable sites that meet technological, environmental, spatial, and safety requirements for the deployment of Small Modular Reactors (SMRs) across the entire territory of the Republic of Slovenia.

The site selection methodology is based on a multi-stage approach comprising suitability analysis, vulnerability assessment, and comparative site adequacy assessment. The methodology ensures transparency in decision-making and establishes a reproducible decision-support model. A key element of the applied methodology is the clear definition of criteria used to exclude unsuitable locations, enable the comparative evaluation of suitable locations, and rank the most appropriate sites.

The model is based exclusively on publicly available datasets; therefore, its accuracy depends on the quality, resolution, and availability of data in formats compatible with GiS-based analysis.

In the first phase, a suitability analysis was conducted using exclusionary criteria. These criteria are derived from IAEA safety guide SSG-35, the technological requirements of the planned nuclear facilities, and established international siting practices in countries such as the United States, Canada, and the United Kingdom. Based on these criteria, areas that do not meet key technological and safety requirements for SMR deployment were excluded from further consideration. The principal technological and safety-related exclusion criteria include the availability of sufficient cooling water resources, appropriate physical terrain characteristics, seismic safety, adequate distance from active faults, and relevant geological conditions.

In the second phase, a vulnerability assessment was performed, encompassing both environmental and spatial vulnerability. Environmental exclusion criteria eliminate water protection zones, flood-prone areas, protected natural areas and Natura 2000 sites, forest reserves, and cultural heritage areas. Spatial and external hazard criteria exclude outstanding landscape areas, proximity to airports, defence-related zones, and gas fields.

In the third phase, a comparative site adequacy assessment was conducted for all locations that met the technological, safety, environmental, and spatial requirements. This phase applied additional criteria, including accessibility to cooling water, feasibility of connection to the electricity transmission network, transport accessibility, geological characteristics, flood risk implications, impacts on nature and cultural heritage, population density, presence of brownfield areas, distance from settlements, and proximity to facilities posing potential external hazards to a nuclear installation. The evaluated locations were compared descriptively and tabular based on the developed GiS models, taking into account the presence of specific criteria and the associated development challenges for each site.

In the final phase, all potentially suitable sites were ranked according to the number and significance of key constraints affecting their overall feasibility. The final outcome of the methodological process is a set of potentially suitable sites, accompanied by the identification of key activities required for their further development.

The presented methodology provides a transparent, reproducible, and professionally grounded framework for subsequent planning stages and serves as a robust basis for strategic decision-making regarding the siting of SMRs in Slovenia.

Keywords: *deployment, methodology, siting, Slovenia, SMR*

Session 4
Regulatory Practice, Licencing, Emergency Preparedness, Safety Culture and Public Relations (RPLEP)

S4-112

Analyses of Nuclear Emergency Preparedness and Response Regulations Between U.S. and South Korea

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Since nuclear industry reached maturity with increased number of nuclear power plants operation and radiological research projects, relevant nuclear regulatory framework has evolved together. Among each cornerstone of nuclear regulation, nuclear emergency preparedness and response plays the key rule for securing workers, public, and environment from release of radioactive materials. Emergency preparedness and response regulation has evolved followed by major historical events. In U.S., for example, importance of emergency preparedness and response was emphasized as thermal power of each nuclear reactor unit increased up to 3 GW. After TMI-2 accident in 1979, concept of emergency planning zone and its sizing methodology was specified in U.S. Chernobyl accident and Fukushima accident have facilitated communications between adjacent countries and enhanced understanding of actual radiological consequences, respectively. In 2014, International Atomic Energy Agency (IAEA) published IAEA GSR Part 7, providing international standard for emergency preparedness and response framework among member states. However, it is inevitable for every member states have different regulations. Although South Korea has adopted regulatory framework from U.S. and IAEA, for example, there is certain inconsistency. Such inconsistency will be the challenges for potential adoption of small modular reactors and other new technologies and establishment of supply chains for them. In this study, therefore, current status of U.S. and South Korea's regulatory framework for nuclear emergency was investigated to analyse gaps and look for improvement. When analysing regulatory framework of both countries, 10 items were considered to systemize gaps between two countries, which were established based on recent regulatory guides of U.S. Nuclear Regulatory Commission. Then, it was discussed how to improve and harmonize current South Korea's regulation in future, with lessons learned from 8 defined issues.

Keywords: *emergency preparedness and response, gap analysis, United States, South Korea*

S4-118

Slovenian Nuclear Safety Administration Foreign Operating Experiences System

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The Slovenian Nuclear Safety Administration (SNSA) has established a structured system for the review and evaluation of foreign operating experience (FOE) to ensure continuous improvement of nuclear and radiation safety. The system provides a systematic approach for the identification, screening, assessment, and regulatory follow-up of operating experience from foreign nuclear installations, international reporting systems, and relevant regulatory bodies. FOE information is collected through multiple channels, including international organizations, bilateral exchanges, vendor communications, and publicly available databases. Events and findings are screened based on predefined safety significance criteria and applicability to domestic facilities. Relevant FOE is subject to in-depth technical review to identify potential safety implications, lessons learned, and the need for regulatory actions. The outcomes of the review process are documented, tracked, and communicated to licensees, with follow-up activities ensuring timely implementation and verification of corrective measures where necessary. This system supports proactive regulatory oversight, strengthens defence-in-depth, and contributes to harmonization with international nuclear safety practices.

Keywords: *foreign operating experience, event review, lessons learned, regulatory oversight*

S4-120

Revision of Siting Criteria and Status of Emergency Planning Zone in Korea: Implications for Small Modular Reactors

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Recently, Republic of Korea revised the siting criteria of nuclear facilities in order to reorganize its regulatory framework governing the siting of nuclear energy utilization facilities. The revised siting criteria notice departs from the conventional deterministic criteria centred on large pressurized water reactors and has been improved into a system that can reflect site characteristics and the design features of nuclear facilities. Its objective is to expand institutional flexibility by considering the introduction of next-generation nuclear technologies. Such changes are considered to be closely related to the feasibility of deploying small modular reactors (SMRs) in Republic of Korea. In contrast, the emergency planning zone (EPZ) continues to maintain a deterministic distance-based concept, typically preserving a maximum radius of up to 30 km.

The current EPZ is defined as a fixed radius regardless of reactor power or design characteristics, and the same range is applied to SMRs without separate criteria. As a result, the safety design features of SMRs, such as a smaller source term, enhanced passive safety systems, and reduced radioactive inventories, are not sufficiently reflected at the institutional level.

Consequently, the EPZ may function in practice as a siting constraint. This situation highlights a regulatory inconsistency between the revised siting criteria and the EPZ framework. While siting regulations are becoming more flexible, the continued applications of a uniform EPZ may limit site diversification and function as a policy barrier to SMR deployment. Internationally, discussions are expanding on applying scalable EPZs for SMRs through risk-informed and performance-based (RIPB) regulatory approaches, reflecting probabilistic safety assessment (PSA) results and reactor design characteristics. In the United States, regulatory discussion and licensing practices for SMRs, such as those reflected in NRC guidance and design certification reviews, have increasingly considered RIPB regulatory approaches for SMRs, with ongoing discussions on the applicability of scalable or reduced EPZs that reflect PSA results and design-specific safety features rather than fixed distanced-based criteria.

This study compares and analyses the revised siting criteria with the current EPZ regulations and examines potential institutional improvements, including the feasibility of introducing RIPB EPZ framework.

Keywords: *Siting Criteria, Emergency Planning Zone, SMR, PSA, Risk-informed, Performance-based*

S4-133

Public Opinion Survey on Nuclear Energy and Radioactive Waste Management 2026

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During the beginning of 2026 the Department of Applied Physics of the Faculty of Electrical Engineering and Computing, University of Zagreb conducted a public opinion survey entitled "Nuclear Energy and Radioactive Waste Management" primarily among student population but also partially grasping general population. The survey is a continuation of previously conducted surveys which enables comparison of participants' positions and analysis of trends. Taking into account the recent political agenda in Croatia, a set of questions aiming at Small Modular Reactors (SMR) and their potential application in Croatia was added to the survey.

Keywords: *public opinion survey, nuclear energy, radioactive waste management, small modular reactor (SMR)*

S4-146

Building Relationship with the Local Community

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Effective public communication and proactive community engagement are critical, yet often underestimated, components of safe and sustainable nuclear projects. The intension of this presentation is to demonstrates how investing in the social fabric is not just a regulatory obligation, but a fundamental pillar of long-term operational success and social responsibility in the nuclear field. As part of establishing the Radioactive Waste Management Centre, the Fund focuses a key part of its long-standing public relations work specifically on building strong relationships with the local community. Building good relations and trust is a long-term process that requires a systematic approach, honesty, and continuity. Trust cannot be established overnight; building it necessitates a considerable investment of effort, work, and dedication. This presentation will outline the activities conducted by the Fund with the local community, explaining how we have structured and tailored them specifically toward community engagement. The activities are designed with careful consideration for the specificities of the region, the needs of the local community, and the goal of optimizing information dissemination. Given the broad scope of local community needs, our activities are directed toward a diverse range of areas, including agriculture, economy, social welfare, healthcare, education, family farms, and more. Furthermore, our local community has specific preferences regarding how it receives information. Consequently, effective information transfer requires utilizing a variety of channels (such as websites, social media, traditional media, print materials, and study visits), as well as a suite of communication tools like information centres, and visual, audio, and video materials. Ultimately, this tailored and multifaceted approach is designed to build the lasting trust and cooperative foundation necessary for the Centre's sustainable operation and integration into the community.

Keywords: *public relations, communication, local community, building trust*

S4-157

Technical Status of EPZ Evaluation for SMRs in Korea

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Global interest in small modular reactors (SMRs) is rapidly increasing, and South Korea is developing an innovative SMR (i-SMR). Based on pressurized water reactor technology, the i-SMR offers various advantages in terms of safety, economic viability, resilience, and flexibility. The establishment of an emergency planning zone (EPZ) is mandatory for the construction and operation of innovative SMRs. Current domestic regulations require adherence to the basic area of EPZ structure comprising the precautionary action zone (PAZ) and urgent protective action planning zone (UPZ), which was established based on existing large light water reactors. This indicates that existing regulatory requirements do not correspond to the power capacity and safety characteristics of SMRs, making it impossible to apply appropriate regulations.

In the United States, 10 CFR 50.160 (Emergency preparedness for SMRs, non-LWRs, and non-power production or utilization facilities) was newly established for SMRs and other nuclear facilities. This regulation provides graded approach that considers the characteristics of individual next-generation nuclear technologies. Additionally, RG 1.242 (Performance-based emergency preparedness for SMRs, non-LWRs, and non-power production or utilization facilities) was published to provide detailed guidance on emergency planning requirements, including methodologies for evaluating emergency planning zones that licensees can follow.

This study organized the background necessary to understand this situation, analysed the gaps with existing domestic regulatory standards, and proposed solutions. For this purpose, the study summarized the power capacity and safety characteristics of i-SMR and analysed the inadequacies and limitations of applying the existing regulatory framework to i-SMR. This study also reviewed existing SMR EPZ assessment methodologies and proposed a draft EPZ assessment method for i-SMR based on the U.S. RG 1.242 approach. It further covered the development of the Radiological Consequence Analysis Program - Emergency Planning Zone (RCAP-EPZ) for evaluating EPZs. This study avoided specifying detailed criteria such as frequency criteria for accident scenario selection or dose criteria for EPZ evaluation, instead describing general methodologies.

It is expected that this study will aid in understanding the current situation and global technological level, and provide insights to bridge regulatory and technical gaps in establishing EPZs for SMRs, both domestically and internationally.

Keywords: *Emergency Planning Zone, Small Modular Reactor, Offsite Consequence Analysis, Dose Assessment, Probabilistic Safety Assessment*

S4-184

Exploring Trends and Patterns in International Event Reporting

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Various international systems were developed for event reporting that enable exchange of operational experience (OE) to improve the safety of nuclear power plants. This paper presents a review of event reports from the International Atomic Energy Agency (IAEA) International Reporting System (IRS) database, focusing on trends and recurring themes relevant to nuclear plant operation and maintenance. Drawing from reported events, the study applies frequency analysis, temporal trending, event grouping, and causal node mapping to explore common causes, safety relevance and significance, interdependencies across systems, plant states, and operational contexts. The approach includes descriptive metrics related to reactor trips, equipment performance degradation and human performance, as well as visual tools that highlight potential relationships between technical, procedural, and human factors. The analysis is structured to allow multiple patterns to emerge - such as persistent vulnerabilities or patterns that cut across systems, causes, or organizational factors. While specific conclusions will depend on the data, the paper aims to contribute to more structured and transferable use of OE information and provide a foundation for developing consistent trending approaches across the industry to be able to achieve higher level of nuclear safety.

Keywords: *Operational experience (OE), Event reporting, International Reporting System (IRS), Nuclear safety trending*

S4-194

The Role of Information Centres in Communication with the Public on Radioactive Waste Management

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For the purpose of informing the public, the Fund has established two Information Centres on radioactive waste, in Dvor and Zagreb. The goal of establishing the Information Centres is to provide comprehensive information to different age groups in an accessible and understandable way, and to involve the public in the project's progress.

By organizing various activities, in the fields of education, health, agriculture, and social services, as well as participating in public events, the number of visitors has increased. The results of a survey of the population in the Municipality of Dvor showed that 80% of respondents had heard of the Information Centre on radioactive waste in Dvor.

At Information Centres, visitors can use virtual reality to tour facility for receiving, classifying, pre-processing, treating, and storing radioactive waste. The content of the Info Centre includes a 3D model of the Centre for the management of radioactive waste under construction with a view of the storage space, and an enlarged view of the concrete tank with barrels of radioactive waste.

The fund regularly posts news about activities held at the Information Centres on its website, www.czrao.hr, and Facebook page.

Keywords: *public relations, communication, local community, building trust, education*

Session 5
Reactor Physics and Nuclear Fuel Cycle (RPNFC)

S5-129

Closing the Loop: Why the Nuclear Fuel Cycle Matters for a Sustainable European Energy System

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Europe is entering a period of rapidly rising electricity demand, driven by electrification, digitalisation, synthetic fuels, and industrial reshoring. At the same time, energy policy is increasingly constrained by land use, material intensity, cost volatility, and long-term environmental impact. In this context, nuclear energy is regaining strategic relevance — not as an ideological choice, but as a response to physical, material, and systemic constraints. This paper examines the Closed Nuclear Fuel Cycle (CFC) as a system-level solution rather than a collection of individual technologies. Drawing on historical experience with fast reactors and fuel recycling in Europe and internationally, this contribution argues that the long-term sustainability of nuclear energy may be a function of scaling the CFC. The scalability of the CFC depends on deliberate development of recycling capacity and fuel-cycle infrastructure. Deployment rate of reactors in a CFC is directly limited by the amount of recycled materials being made available, and once the reactors are operating the scalability is a function of "compounding interest". Every reactor can produce a certain amount of fissile material per year, and new reactors can be started only as more fissile material becomes available. The earlier the recycling is started and the higher capacity of recycling, the greater the scalability for reactor capacity would be. Using Europe as a worked example, the presentation explores the scale, sequencing, and governance implications of transitioning from a once-through fuel cycle to a mature CFC based on fast-spectrum reactors. The analysis highlights that the most significant rate-limiting factor in such a transition may initially be the pace at which fuel recycling can be industrially scaled under robust safeguards and regulatory oversight. Expansion rate of an FBR fleet is a function of available plutonium.

Non-proliferation, cross-border material governance, and institutional continuity are treated as foundational design constraints, not secondary considerations. Rather than proposing a specific technology roadmap or political programme, the contribution frames the closed fuel cycle as a boundary condition imposed by physics, resource efficiency, and land-use constraints, if deep decarbonisation is to be achieved without large-scale industrialisation of natural landscapes. The closed fuel cycle is presented not as a radical departure from existing nuclear practice, but as the completion of an energy system originally conceived to operate in this manner.

Keywords: *Closed Fuel Cycle, Recycling of Nuclear Fuel, Reprocessing, Safeguards, Non-Proliferation*

S5-132

Calculation of PWR Reactor Xenon Poisoning Using the Wigner-Seitz Approximation and Physics-Informed Neural Networks

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During the PWR reactor operation, fission products accumulate in the nuclear fuel giving an increase in the absorption of thermal neutrons, which reduces the reactor reactivity. This phenomenon, known as reactor poisoning, is primarily associated with the xenon ^{135}Xe and samarium ^{149}Sm isotopes, which reach their maximum concentration relatively quickly during the reactor operation. The biggest contribution to poisoning comes from the isotope ^{135}Xe due to its extremely large absorption cross section for thermal neutrons (about 3 million barns), so knowing its temporal concentration is essential for calculating the change in reactor reactivity. In order to assess the influence of a particular radionuclide on the reactor's reactivity, it is necessary to solve its Bateman equations, which describe the balance of the processes of creation and disappearance of nuclides in the neutron field. The analytical solution to ^{135}Xe temporal concentration includes the thermal neutron utilization factor (*f*-factor), which will be explicitly modelled in this paper. The heterogeneous unit fuel cell will be modelled using the neutron diffusion theory in cylindrical geometry, within the Wigner-Seitz approximation. The obtained analytical and numerical results of the *f*-factor, ^{135}Xe concentrations, and resulting changes in reactor reactivity will serve as input data for training physics-informed neural network (PINN) algorithms. This test-case from reactor physics will examine PINNs applicability and level of their numerical predictive accuracy.

Keywords: *xenon, PWR reactor poisoning, thermal neutron utilization factor, Wigner-Seitz approximation, physics-informed neural networks*

S5-139

Extension of the WIMSD-5B Code for Multi-Zone Fuel Modelling Using Geometry-Consistent Dancoff Factors

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Deterministic reactor-physics codes remain important tools for routine reactor analysis, burnup calculations, and few-group constant generation. Within this class, the WIMS family of codes continues to be of practical interest, but its applicability must be extended when geometries fall outside its original standard scope. One such case is the annular, multi-zone fuel used in the RB and RA research reactors in Vinča. For this geometry, the standard resonance treatment in WIMSD-5B does not preserve the correct neutron escape-and-return behaviour of cylindrical shell-type fuel, which leads to an overestimation of the corresponding Dancoff factor and affects resonance self-shielding and the resulting group-averaged cross sections. In this work, a procedure for extending the applicability of WIMSD-5B to annular multi-zone fuel analysis is developed using geometry-consistent Dancoff factors. The procedure is based on the VEGA2DAN sequence of the VEGA-2 lattice-physics code. VEGA2DAN determines effective Dancoff factors on the basis of the equivalence principle by preserving the absorption self-shielded cross section in a selected energy range between the real heterogeneous annular geometry and an equivalent geometry used in the resonance treatment. The resulting Dancoff factors were supplied to WIMSD-5B without modification of its core solution algorithms. The methodology was demonstrated on the RA reactor fuel element, for which both a detailed three-dimensional model and an equivalent deterministic one-dimensional multizone model were constructed. Burnup calculations were benchmarked against VEGA-2/ORIGEN-2.2, TRITON from SCALE-5.1, MCNP-5/ORIGEN-2.2, and MCNPX-2.7/CINDER'90, the last of these being used as the principal reference methodology. Calculations were performed with ENDF/B-VI.6 and ENDF/B-VII.0 data. The modified WIMSD-5B approach reproduced the burnup dependence of the infinite multiplication factor with acceptable deviations for the present heavy-water application, while very good agreement was obtained for the evolution of ^{235}U concentration. The ^{134}Cs / ^{137}Cs concentration ratio proved to be a more sensitive indicator of resonance-treatment quality, confirming that the remaining limitations of the equivalence-based model are most visible in isotopic quantities influenced by resonance interference effects.

Keywords: *resonance absorption, Dancoff factor, cylindrical shell-type fuel, WIMSD-5B, MCNP-5*

S5-141

Development and Optimization of Geometric Models of a HPGe Detector for Numerical Efficiency Calculation

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This paper presents the development, optimization, and validation of a numerical model of the coaxial HPGe detector ORTEC GEMC100 for the determination of full-energy peak efficiency required in gamma-spectrometric activity measurements. The model was developed to support the needs of "Nuclear Facilities of Serbia", LLC during the decommissioning of hangars H1 and H2 and the RA research reactor, where complex and non-standard source geometries often make direct experimental efficiency determination impractical and, in some cases, impossible. Particular emphasis was placed on achieving a level of reliability such that the uncertainty of simulated efficiencies is comparable to the measurement uncertainty of certified radioactive calibration sources. The detector modelling was based on crystal geometry and construction parameters from the manufacturer's technical documentation, and a detailed three-dimensional model was implemented in the MCNP-5 Monte Carlo code. Model optimization was performed through iterative adjustment of dead-layer thicknesses and related structural parameters, using measurements with certified point sources, a certified 1500 cm^3 Marinelli beaker, and a certified CBSS2 reference source. Additional cross-verification was carried out by comparing efficiencies calculated with MCNP-5 and FLUKA 4-4.1. The validated detector model was then used as input for the EFFTRAN software in an operational test case, and the obtained results were compared with full Monte Carlo calculations and with the ISOCS software for the Canberra GX5020 detector. The results showed good agreement between simulated and experimental data, as well as close consistency among the different computational approaches. The developed methodology therefore provides a reliable basis for practical numerical efficiency determination and a foundation for future software solutions applicable to semiconductor and scintillation detectors of various types and manufacturers.

Keywords: *HPGe detectors, detection efficiency, numerical calibration, MCNPX-2.7, FLUKA 4-4.1*

S5-142

Shielding Analysis of a Reinforced Concrete Container for Conditioning Low- and Intermediate-level Radioactive Waste from Nuclear Power Plant Krško

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This paper presents a shielding analysis of the Croatian design of reinforced concrete container (RCC) using MCNP code. The structural design of the container was developed by the University of Zagreb Faculty of Civil Engineering. The RCC is intended for low and intermediate level radioactive waste storage in new Radioactive waste management centre Čerkezovac. The basic RCC calculation is performed according to container design specifications. The RCC contains four metal drums with radioactive waste, which are surrounded by grout mixture. The dose rates were tallied at multiple locations around and at the surface of the RCC to check fulfilment of regulatory requirements defined in the document IAEA SSR-6 "Regulations for the Safe Transport of Radioactive Material" Revision 1. The regulatory limits are, in terms of dose rates, 2 mSv/h at the RCC surface and 0.1 mSv/h at 2 m away from the RCC surface. The calculations were performed for three types of radioactive sources derived from characteristics of NPP Krško waste (maximum resin source, maximum non-resin source, and average radioactive source) and described in terms of source intensity and spectrum. The additional sensitivity analyses were aimed to address potential deviations in concrete and grout density, and deviations in RCC wall dimensions from the original cask design. To validate MCNP results, MAVRIC module from SCALE 6.2.4/6.3.1 was used.

Keywords: *reinforced concrete container, dose rate, shielding, MCNP, SCALE*

S5-145

Optimization of Spent Fuel Assemblies Arrangement in Dry Storage Casks using Mixed Integer Programming and Heuristic Methods

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Spent fuel dry storage (SFDS) project in Nuclear Power Plant (NPP) Krško covers transfer of spent fuel from spent fuel pool to Dry storage building in four loading campaigns. The first loading campaign was performed in 2023 according to the fuel loading plan proposed by the casks vendor Holtec International and the rest of the loading campaigns are expected to start in 2028, 2038 and 2048, respectively. In this paper, the arrangement of available spent fuel assemblies (SFA) in SFDS HI-STORM FW storage casks for future loading campaigns is proposed. In campaign two 16 casks will be filled, in campaign three 12 casks and in campaign four 18 casks. There is also an option that the last two campaigns might be merged into one, so in that case, there would be in total 3 loading campaigns and in the last campaign 30 casks would be filled. Each storage cask can accept up to 37 SFAs, divided into three spatial regions according to heat rate limit (maximum cladding temperatures during hypothetical accident) for each region. The proposed arrangement for each of the following campaigns is a result of optimization process using Mixed Integer Programming and heuristic optimization methods, both capable to comply with required technical and logistic constraints. The cost function can be either minimization or maximization of SFAs decay heat in casks depending on of spent fuel pool management strategy, but it can also be defined as a uniform decay heat distribution among casks to avoid potential overheating. The SFAs decay heat used in optimization process was calculated using Origen-S module from SCALE code package using real operating history and cooling time of NPP Krško SFAs. The optimization was performed considering the number of SFAs in each campaign (both four and three loading campaigns considered), minimum cooling period of 5 years in spent fuel pool, plant operation until year 2043, region-wise heat rate casks limits, and additional constrains on cask locations allowed to accept SFAs with inserts.

Keywords: *spent fuel assembly, spent fuel dry storage, HI-STORM FW, mixed integer programming, decay heat*

S5-164

Criticality Sensitivity Analysis of a Generic Spent Nuclear Fuel Disposal Cask

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Criticality safety is one of the key challenges in the design of disposal concepts for spent nuclear fuel. This work presents a criticality sensitivity analysis of a generic spent nuclear fuel disposal cask specification proposed within the Eurad2 WP17 project (CSFD – Criticality Safety for Final Disposal). The study aims to quantify the impact of relevant design and degradation parameters on the effective neutron multiplication factor and to provide input for the derivation of spent fuel loading curves. Fresh fuel configurations are considered as a reference, including uranium enrichments of 2%, 3%, 4%, and 5% and two common fuel assembly types (16×16 and 17×17). Several criticality-relevant cases are investigated, focusing on the positioning of fuel assemblies within the cask boxes, variations in water density, and changes in the thickness of structural material between assembly compartments. In addition, long-term degradation effects are addressed by analysing stainless steel insert corrosion and the formation of a magnetite layer, which may influence neutron moderation and reflection behaviour over disposal timescales. All calculations are performed using the Serpent2 Monte Carlo code with nuclear data libraries ENDF/B-VII.1 and ENDF/B-VIII.1, allowing for comparison of cross-section impacts on sensitivity results. The outcomes of this work contribute to a better understanding of parameter-driven uncertainties in disposal cask criticality assessments and support the development of robust safety margins for final repository concepts.

Keywords: *spent nuclear fuel disposal, criticality safety, sensitivity study*

S5-169

Application of Composite Buildup Factors in Point-Kernel Method for Laminated Gamma Shields

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In practical applications many shields against gamma radiation are often laminated (or composite), using layers of different materials. Such composite shields introduce difficulties for calculating buildup factors, since change in angular and energy distribution of gamma-rays occur during the passage through intervening layers. Additional complication comes from interchangeable materials, since gamma-rays in every layer depend on previous layers, so this memory effect has an implication on layer ordering inside the shield. The accumulated empirical data over history has served as a rich database for introducing several approximations to composite buildup factors, such as methods of Goldstein, Blizard, Broder, and Bowman-Trubey. These methods proved quite useful for applications in gamma-ray shielding when using the deterministic point-kernel (PK) methods. This paper presents underlying analytical derivations and implementation of such methods in our existing PK program for gamma shielding, which is still in a development phase. The selected test-cases were focused on calculating the buildup flux on a point detector, originating from a slab gamma source shielded by a slab shield. The volumetric source has an explicit photon self-absorption, with a constant or exponential gamma-ray emission rate, while both slabs can have variable thickness and shielding materials. The obtained sensitivity study of PK deterministic results was verified using the stochastic Monte Carlo method, to compare similarities and differences in gamma flux components (uncollided and collided) and composite buildup factors.

Keywords: *buildup factor, point-kernel, composite shield, gamma radiation, Monte Carlo method*

S5-170

Severe Accident Gamma Dose Mapping of PWR Nuclear Island Using FW-CADIS Methodology and Contributon Flux

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The hybrid deterministic-stochastic shielding methodology of FW-CADIS was used to find global gamma dose distribution in case of station blackout (SBO) with small loss of coolant accident (LOCA) in a realistic PWR nuclear island model. The MAVRIC shielding sequence of SCALE6.2.4 program package is controlling the hybrid FW-CADIS methodology that uses deterministic forward-adjoint flux solution over a large computational mesh as a means to produce variance reduction (VR) parameters for accelerating the final Monte Carlo (MC) simulation. Such shielding problems, with massive flux attenuation by many orders of magnitude, can be solved only by an efficient computational workflow. This paper presents numerical trade-offs and influence of discrete ordinates (SN) solution on global MC convergence via space-energy VR parameters (importance map and biased source) that closely work in a tandem. In this way it is possible to prepare a rather detailed and large MC model to analyse gamma flux attenuation from the PWR containment interior to external auxiliary buildings in case of hypothetical SBO accident. The accidental source was calculated using RADTRAD3.03 code and prepared for SCALE6.2.4 by ORIGEN2.2 code, representing gaseous effluents with uniform distribution over all air-regions inside the containment. The source corresponding to activity 2h after the SBO accident gives largest gamma dose rate in auxiliary building so it was used for subsequent MAVRIC calculations. The resulting gamma flux distribution was examined through different sections and compartments to pin-point possible streaming paths important for radiological assessment. MAVRIC auxiliary routines were used for folding forward and adjoint multigroup fluxes into a normalized contributon flux, giving an extra insight how the global response propagates throughout the computational phase space. Special attention was given to gamma dose rates in auxiliary building rooms represented by a forward-weighted adjoint source ensuring uniform particle distribution in those locations of interest. Finally, the obtained gamma dose rates were compared to natural background radiation levels to visualize areas of occupational exclusion.

Keywords: *FW-CADIS, hybrid shielding, Monte Carlo method, gamma dose, severe accident*

S5-175

A Stylized 3D PRISM-Based Nitride Fuel Configuration Benchmark Problem

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A set of 3D sodium-cooled fast reactor benchmark problems based on the General Electric Power Reactor Innovative Small Module (PRISM) for a Nitride fuel variation of the design is developed in this paper. The configurations include an uncontrolled case with all control rods withdrawn, a controlled case with all control rods inserted, and a critical case with some control rods inserted. The benchmark problems retain detailed geometric and material information of the Nitride fuel, liquid sodium coolant, duct gaps, wire spacers, Boron Carbide (B_4C) absorber, stainless steel alloy HT9 cladding for uncontrolled, critical, and controlled assembly configurations, and faithfully adheres to those design specifications that are available publicly.

Keywords: *Benchmark, Nitride Fuel, Sodium, PRISM Reactor*

S5-182

FHR Core Monte Carlo Calculation using Serpent2 Code

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The work presents the steady-state neutronic results of the FHR (Fluoride-Salt-Cooled High-Temperature Reactor) core Monte Carlo calculation. The analyses were performed using a full-core model based on a modified ORNL AHTR reference design as described in OECD NEA FHR/AHTR Benchmark – Phase II-A. Principal modifications include adjustments to the active core height and a reduced number of fuel elements, while the fuel element geometry and material compositions remain consistent with the specifications defined in FHR Benchmark Phase I-C. The FHR core comprises hexagonal fuel elements arranged with 120° rotational symmetry. The fuel has a nominal enrichment of 9.0 wt% U-235 and consists of TRISO particles embedded in graphite plates (referred to as "planks"). FLiBe is utilized as the primary coolant, while graphite serves as both the structural and moderator material. The reactor vessel is modeled using two distinct structural alloys of specified dimensions. Radially, the active core is surrounded by a single ring of replaceable hexagonal reflector elements. These elements, as well as the central core element, contain a central coolant channel. Beyond this region, fixed radial reflector elements – composed entirely of graphite – are modeled with the same external dimensions as the fuel elements. Axially, each fuel element is divided into multiple regions: a bottom axial reflector, a lower non-fueled extruded region, the active fuel region containing TRISO particles, an upper non-fueled extruded region, and a top axial reflector. The calculation assumes uniform temperature and material density distributions throughout the model. The calculated results are multiplication factor, 3D spatial distributions of fission rate density and 3-group neutron flux. The core neutron spectrum using SCALE 252-group energy structure was calculated too. The influence of number of generations and particle histories on spatial distribution of flux uncertainties was shown. The flux was tallied at fuel assembly level and using prescribed 3D Cartesian grid. The results were checked from the point of view of axial (both core and fuel assembly) and radial flux symmetry. Spatial and energy collapsing of flux results was done both externally using data processing and internally using different sampling criteria in detector definitions. All calculations are performed using the Monte Carlo neutron transport code Serpent 2.2.3.

Keywords: *FHR core calculation, Monte Carlo criticality, Serpent2 code*

Session 6
Severe Accident Analyses and Risk Assessment (SAARA)

S6-105

Optimizing the Number of Gates Representing a FTA k/n Operator on a Quantum Computer

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The classic static fault tree analysis is a deductive method based on Boolean logic widely used for reliability and safety analysis. The method is based on the application of logical operators to show the relationship between component failures that lead to intermediate or top-level adverse events. At the lowest level, it is assumed that the events are basic and that the probability of their occurrence is fixed according to the Bernoulli distribution, for instance, the probability of a failure occurring for a component. Recently, it has been shown that such events can be successfully simulated by means of qubits and that events on middle and highest levels can be written by using quantum gates that simulate logic operators, i.e., can be written utilizing a quantum fault tree. The quantitative and qualitative analysis of the static fault tree can then be simulated on that record aided by a quantum computer. In this article, in addition to the standard set of logical operators (and, or, not), we further explore the static fault tree record that contains the commonly used k/n logical operator. Although the k/n operator can be written employing a combination of a standard set of logical operators, however, such notation is usually not optimal for implementation on a quantum computer, except for the simplest cases. Therefore, in this paper we present a model for optimizing the number of quantum gates in a quantum circuit record for events that are described in a static fault tree by the k/n logic operator. In addition, in the given example of such a static fault tree consisting of six basic events and four derived events we closely present the calculation of the top event probability and the determination of all state vectors in the quantum fault tree that represent minimal cut-sets of the latter. Eventually, we analyse the usage and scaling of resources in a quantum computer and additionally provide a summary of practical limitations in the application of a hybrid static fault tree recording strategy on potential quantum computers in the near future.

Keywords: *Probabilistic Safety Assessment (PSA), Fault Tree Analysis (FTA), Quantum Fault Tree (QFT)*

S6-119

Risk-Informed SSC Categorization for an Innovative SMR under NEI 00-04 (10 CFR 50.69)

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Innovative small modular reactors (i-SMRs) being developed in Korea employ extensive passive safety features and modular/shared configurations. These characteristics can limit the effectiveness of deterministic-only safety classification when proportional quality assurance (QA) and resource allocation are desired while preserving defence-in-depth. This abstract proposes a risk-informed SSC categorization methodology for a Korean i-SMR explicitly based on NEI 00-04, the implementation guidance for 10 CFR 50.69, and defines an SDA-oriented procedure and deliverables. Consistent with NEI 00-04, an SSC is treated as safety-significant if it is judged important from any of four perspectives: risk significance, defence-in-depth (DiD), risk sensitivity, and the Integrated Decision Panel (IDP). The workflow follows eight steps: (1) collect plant/design inputs and review PRA technical adequacy; (2) engineering evaluation (system boundaries and functional/failure mapping); (3) risk-significance evaluation using PRA importance measures; (4) DiD evaluation; (5) preliminary functional categorization; (6) sensitivity analyses (e.g., human error, common-cause failure (CCF), and maintenance unavailability variations); (7) IDP review and approval; and (8) final categorization into RISC-1 through RISC-4 with controlled documentation. SMR-specific considerations are addressed, including limited PRA scope and data maturity at the design stage, inter-module dependencies and shared-system interactions, CCF impacts, and the need to treat low-probability passive failure modes appropriately. Deliverables include a classification basis report, traceability matrices linking SSC functions to PRA and deterministic bases, and interfaces to configuration/change management to maintain consistency as the design evolves.

Keywords: *Innovative SMR; NEI 00-04; 10 CFR 50.69; RISC-1–4; Risk-informed; SSC categorization; IDP; PRA*

S6-143

Feasibility Assessment of Structural and Component Enhancements at Generation II Pressurized Water Reactors for Implementing External Reactor Vessel Cooling (ERVC) in Severe Accident Mitigation

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This study evaluates the feasibility of external reactor vessel cooling (ERVC) improvement as a severe accident mitigation strategy at Generation II pressurized water reactor not originally designed for in-vessel corium retention (IVR). To obtain more realistic results, the assessment is based on the design of NPP Krško, which has already demonstrated the feasibility of ERVC implementation with reasonable confidence in success. The assessment integrates theoretical and numerical analyses, plant specific probabilistic and deterministic (MAAP 5.03 code) simulations, and a review of existing accident management procedures, including recent scientific and industry insights related to ERVC improvement. Key factors including reactor cavity geometry, critical time windows for dominant plant damage states, thermal power characteristics, and the impact of reflective insulation are examined to determine ERVC performance limits and potential enhancement pathways. The study identifies general measures which might improve ERVC capabilities, such as improvements to cavity flooding, the applicability of additional ex-vessel cooling features, thermal insulation upgrades, and the influence and compatibility of other concurrent mitigation strategies, together with accident management procedural enhancements. Findings indicate that with targeted structural and procedural upgrades, ERVC can be upgraded to a more viable and robust component of severe accident management, strengthening containment integrity and overall defence in depth against extreme external events. The study also identifies the need for further research to quantify such enhancements by defining performance indicators for ERVC in terms of reducing the probability of radioactive material release into the environment and the associated radiological consequences.

Keywords: *in-vessel corium retention, external reactor vessel cooling, severe accident management, pressurized water reactors*

S6-149

Detailed Circuit Analyses for Fire Probabilistic Risk Assessment (NUREG/CR-6850, EPRI 1011989)

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As part of the Fire Probabilistic Risk Assessment (Fire PRA) detailed circuit analyses were performed for the event of a fire occurring on equipment declared as Safe Shutdown Equipment List (SSEL). Fire PRA methods have been used in the Individual Plant Examinations of External Event (IPEEE) program to facilitate a nuclear power plant examination for vulnerabilities. According to NUREG 6850 these studies address the full breadth of Fire PRA technical issues for power operations and include consideration of large early release frequency.

The article represents complete circuits analysis method in according to NUREG-6850 by three tasks:

Task 3: Cable selection (selection Fire PRA cables to dedicated SSEL equipment, which failure can affect their operation), Task 9: Detailed circuit analyse of SSEL equipment, Task 10: Circuit failure mode likelihood analysis.

First phase of circuit analysis was approach of selecting Fire PRA cables. In all of our cases, it was advantageous to perform all of Task 9 (detailed circuit failure analysis) cables within Task 3. The degree to which Task 3 and Task 9 are combined is highly dependent on plant-specific. Second phase of analyse was to conduct a more detailed analysis of circuit operation and functionality to determine equipment responses to specific cable failure modes. These relationships were then used to further refine the original cable selection by screening out cables that cannot prevent a component from completing its credited function. Second phase contains the following key elements: Determine the component response to postulated conductor/cable failure modes, Screen out cables that do not impact the ability of a component to complete its credited function. Third phase was presenting Task 10, which is estimating the probability of hot short cable failure modes of interest, which in turn can be correlated to specific component failure modes. Within that phase we estimate specific cable failure modes associated with fire-induced cable damage. Intention was to provide a probabilistic assessment of the likelihood that a cable will experience one or more specific failure modes (e.g., short-to-ground, intra-cable conductor-to-conductor short, inter-cable conductor-to-conductor short, etc.).

Keywords: *Fire PSA, circuit analyses, cable selection*

S6-181

Verification and Validation of GOTHIC 8.5(QA) with Severe Accident Containment Experiments

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Increase in energy consumption requires increasing demand for energy sources. On the other hand, greenhouse gas emissions and climate change make it challenging to find reliable and sustainable energy solutions. Nuclear energy has high potential to meet energy needs without CO_2 generation. However, nuclear energy also comes with challenges, and one of the biggest is safety. There are a lot of reactor types. But regardless of reactor type, whether conventional large-scale plants or newer designs such as Micro Modular Reactors (MMRs) and Small Modular Reactors (SMRs), containment performance plays a central role as the final protective layer against radioactive material release and in overall plant safety. Although advance reactor concepts aim to enhance inherent and passive safety features, the need for reliable containment analysis and validation remains essential for both existing and emerging technologies. GOTHIC 8.5(QA) is a comprehensive and up-to-date reactor containment analysis code that performs advanced thermal-hydraulic (T/H) calculations for deterministic safety analyses, probabilistic safety assessment, and long-term operation support of nuclear power plants. This paper presents the validation and verification (V&V) of GOTHIC 8.5(QA) against selected large-scale experimental benchmarks: NUPEC M-7-1, PANDA ST3_2, and TOSQAN tests. These experiments represent important containment phenomena during a severe accident, such as hydrogen behaviour, gas mixing, stratification, and steam condensation. Detailed nodalization models were developed for each facility, including review of boundary conditions and geometrical fidelity to evaluate modelling quality. Simulation results were compared with experimental data in terms of pressure, temperature, and gas concentration evolution. Particular attention was given to hydrogen behaviour and distribution, transient evolution, peak values, and long-term trends.

The comparison shows generally good agreement between simulation and experimental results. Some deviations were observed in local mixing and stratification prediction, which require further sensitivity studies through mesh refinement and modelling adjustments. The study confirms that GOTHIC 8.5(QA) can reproduce key containment thermal-hydraulic phenomena during severe accidents with acceptable accuracy for safety analysis applications.

Keywords: *thermal-hydraulics, severe accident, GOTHIC 8.5(QA), NUPEC, TOSQAN, PANDA*

S6-187

Thermohydraulic Influence on the Severe Accident Progression in a Small Modular Reactor

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System dynamics in the small modular reactor (SMR) is determined by the large amount of water in the reactor vessel and the interaction with the small volume of the containment. The amount of decay heat is also smaller due to the low nominal power. If a significant pressure reduction is ensured in the initial phase of the accident, the core will be submerged for a long period of time. Fluid loss through the pressurizer valves will not significantly reduce the reactor coolant system (RCS) inventory, but coolant leakage through a potential rupture site will be minimal due to the low pressure in the reactor vessel. The analyses conducted in this study will focus on the influence of thermohydraulic boundary conditions on the heat removal capability in critical situations. The influence of safety systems will be neglected, or significantly limited, and the initial conditions will necessarily be conservative. The probability of a severe accident in a small modular reactor is low, but the accident should still be taken into consideration in order to provide a basis for the preparation of technical documentation, e.g. severe accident management guidelines, or emergency planning zones. The calculations will be performed with the ASYST code, which combines a detailed thermohydraulic model in design basis conditions and a mechanistic approach in assessing the progression of a severe accident. The core quench tracking, hydrogen production, core melt, molten pool progression and other main severe accident phenomena will be studied. The reactor system chosen is IRIS, a classic example of a light water SMR with an integral reactor vessel and a small containment with built-in passive safety systems. The numerical model is complex and enables a realistic simulation of the power plant behaviour.

Keywords: *small modular reactor, safety analysis, IRIS, severe accident, ASYST, passive safety systems*

S6-188

Determining LOCA Initiator Categories for a PSA of a New Reactor Design

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When developing Probabilistic Safety Assessment (PSA) for a new reactor design, initiators are often categorized on the basis of generic criteria used for the existing nuclear power plants such as large Pressurized Water Reactors (PWRs). However, when a new NPP design differs significantly from conventional large PWRs, the absence of plant-specific deterministic justification for the adopted LOCA break size thresholds (large, medium, small) introduces uncertainty in accident sequence modelling and success criteria definition as well as potential incorrect definition of the initiating events frequencies. A deterministic, design-specific LOCA categorization should be developed with the objective of establishing break size ranges that reflect the actual response and functional requirements for the specific plant design, thereby improving the technical credibility of the PSA event tree and fault tree models.

In the following paper, a LOCA break size categorization methodology will be shown, along with dedicated best-estimate deterministic simulation models qualified for the specific NPP design in order to substantiate key assumptions and obtained results. The methodology comprises two main steps: (1) definition of LOCA categories in terms of required system functions for accident mitigation, including short-term core cooling and long-term recirculation; and (2) performance of selected plant-specific thermal-hydraulic analyses to determine representative break size ranges and associated phenomena that can be mitigated by the same combinations of systems. This categorization process follows internationally recognized PSA guidance and standards, including IAEA SSG-3, ASME PRA Standard, IAEA TECDOC-1804, and NRC Regulatory Guide 1.200, which emphasize grouping of initiating events based on similarity of accident progression, plant response, and success criteria of mitigating systems. The resulting LOCA categorization, spanning from Double-Ended Guillotine Break (DEGB) to Very Small LOCA, provides a technically justified basis for identifying the systems required for mitigation of each break size range and for subsequent defining of consistent success criteria within the PSA framework. This methodology aims at enhancing the reliability of accident sequence analysis and supports risk-informed evaluation of plant safety.

Keywords: *Human Reliability Analysis, Large Language Models, Co-analysis*

Session 7
Radioactive Waste Management and Decommissioning, Radiation Hazard and Protection (RWMD)

S7-109

Testing the Response of Several Dosimeters to X-Ray Radiation

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When working with radioactive sources in hospitals, nuclear power plants or research facilities, in order to prevent individuals from being overexposed to radiation, a precise measurement of radiation is of crucial importance. For this reason, there are a number of available dosimeters which are regularly used in those facilities. Since these devices can differ in precision and in the way they detect radiation, it is useful to compare the responses of different dosimeters when exposed to radiation beams. In this paper, we present the results we obtained by testing the responses of several commercially available dosimeters to different beams of X-ray radiation. For each dosimeter, we performed two measurements. Firstly, we irradiated the 1 L spherical ionization chamber, located at the distance of 2 meters from the radioactive source, with beams N-25, N-30, N-40, N-60, N-80, N-100, N-120, N-150 and N-200, with the current being kept at 10 mA. For each beam, the ionization chamber was irradiated until the measured H*(10) dose was approximately 1 mSv. After that, the dosimeters were set at the same distance from the source as the ionization chamber and irradiated with the same beams and for the same amount of time as the chamber. In the second set of measurements, we measured the dependence of the dosimeter response on the dose rate. For this set, the ionization chamber was irradiated by the RQR-8 beam at 100 kV tube potential, while the current was varied, so that the dose rate would change. For each current, the chamber was irradiated until the measured H*(10) dose was again approximately 1 mSv. The dosimeters were then irradiated for the same set of currents and for the same amount of time for each current value. For each dosimeter, the results of the first set of measurements were normalized with respect to the response obtained for the N-100 beam and plotted as a function of the tube potential. For the second set of measurements, the responses of dosimeters were plotted as a function of dose rates.

Keywords: *dosimeters, ionization chamber, X-ray beams, H*(10), dose rate*

S7-111

Segmented Compton Camera with Dual Perpendicular Silicon Photomultiplier Read-Out for 4π Gamma Ray Imaging

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Compton Cameras (CC) are gamma ray imaging systems exploiting the principles of the Compton scattering process for reconstruction of the location of the radioactive sources. They are used in a wide range of fields, including medical diagnostics, environmental monitoring, and homeland security applications such as mapping radiological and post-nuclear contamination. Conventional CC devices employ two layers of detectors-incident gamma rays undergo Compton scattering in the first layer and are subsequently photoelectrically absorbed in the second, giving information about interaction positions and energy deposits at the interaction sites. While these two-layer designs are capable of performing source imaging, they have a limited field of view and low detection efficiency, so they require manual or mechanical rotation systems for useful imaging, as well as long acquisition times to obtain images of sufficient quality.

We propose a novel design of a CC for imaging in 4π with high detection efficiency, utilizing high luminosity scintillating Gadolinium Aluminum Gallium Garnet doped with Cerium (GAGG(Ce)-HL) crystals and silicon photomultipliers (SiPMs). The CC consists of 64 GAGG(Ce)-HL cubical voxels, arranged in four layers of 4x4 crystals. Their light output is detected from two sides, with 4x4 SiPMs positioned perpendicularly to each other. In between the layers are optical reflectors, which contain the optical photons in planes perpendicular to the SiPMs, ensuring the maximum signal is collected, while lowering the probability of inter-crystal leakage. This design offers a depth-of-interaction (DOI) capability and the possibility to use each voxel as a scatterer and an absorber, enabling the 4π imaging capacity. Two system configurations are considered: one employing crystals of 3×3×3 mm³ with a 3.2 mm pitch, and another with 6×6×6 mm³ crystals and a 6.2 mm pitch, both using a one-to-one SiPM readout scheme. Monte Carlo simulations were performed in Geant4 to evaluate the performance of the proposed CC.

We will present results on detection efficiency for point sources at relevant gamma-ray energies and source-to-detector distances, as well as angular resolution and sensitivity to Compton events. We will show preliminary image reconstruction results for a Cs-137 point source, demonstrating the potential of the proposed system for nuclear imaging applications.

Keywords: *Compton Camera, gamma imaging, Monte Carlo simulations, dual perpendicular read-out, silicon photomultipliers*

S7-128

Selection of the Final and Backup Sites for the Czech Deep Geological Repository Including HLW and ILW Disposal

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For deep geological repository the Czech Republic is in process of selecting primary and backup site. Currently there are four sites Březový Potok, Horka, Hrádek, and Janoch all in crystalline rock being considered. The selection process for the final and backup sites will be completed by 2030 in accordance with the strategic plans of the Czech Republic and international requirements concerning ensuring the long-term safety of the repository.

Also, there is consideration that part of Deep geological Repository will not be used for SNF but for HLW and ILW. Two main concepts for this section are being developed i.e. silos and caverns. This section is planned to be separated form SNF section by faultline so that the engineering barriers in SNF section will not be affected by filling material (concreate). Not only the underground is being developed but also the above ground area is being architectonical processed so that its impact on surrounding will be as low as possible.

In 2025, the following projects for the preparation of a DGR were completed:

- Proposal for primary and secondary locations for surface facilities at four sites (Application of the Methodology for the location of surface facilities at individual sites).
- Optimization and conceptual evaluation of technical solutions for the storage of other RAW in a DGR.
- Verification of the required size of rock blocks in the four sites in view of the updated (larger volume) inventory of RAW and SNF.

Current projects for the preparation of a DGR:

- Study of the technical and architectural design of surface facilities in the four sites (Proposal for the technical and architectural design of the surface facility on the area according to the SÚRAO decision).
- Participation in expert groups dealing with the evaluation of the parameters of the mechanical and physical properties of rocks and the size of the underground part of the DGR necessary for the selection of the final and backup sites.
- Preparation of general conceptual documents for the DGR project (proposal for a safety concept).
- Handling of SNF and RAW in the SNF and RAW reception and transfer facility and at the storage horizon, optimization of the hot cell and transfer hub.
- Feasibility study.

Keywords: *Deep Geological Repository (DGR), Decommissioning, Radioactive Waste (RAW), Spent Nuclear Fuel (SNF)*

S7-130

The Study on the Solidification of Radioactive Zeolite Waste Using Alkaline Active Agents for Final Disposal

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This study analysed the compressive strength characteristics of alkaline activator–based mock solidified waste forms of radioactive zeolite and evaluated their potential suitability for final disposal. In heavy-water nuclear power plants, steam recovery systems are used to capture heavy-water vapor released from reactor systems and to limit tritium emissions to the atmosphere, with zeolite commonly employed as the desiccant. During operation, zeolite can adsorb tritium (^{3}H) and radiocarbon (^{14}C) in the forms of HTO and $^{14}CO_2$, as well as gamma-emitting radionuclides released under fuel damage conditions. Without proper solidification and disposal, contaminated zeolite may cause secondary contamination through airborne dispersion or radionuclide leaching. However, a definitive disposal strategy for spent zeolite has not yet been established, and although a portion of the adsorbed ^{3}H and ^{14}C can be removed by pressurization and thermal treatment, gamma-emitting radionuclides remain, necessitating solidification into a monolithic waste form.

In this study, zeolite was combined with alkaline activators to form an aluminosilicate-based matrix. Partial dissolution of the original Si–Al framework during mixing promoted the formation of a new aluminosilicate network, while previously adsorbed radionuclides were concurrently immobilized within the geopolymer structure. Laboratory-scale mock cylindrical waste specimens with a diameter-to-height ratio of 2 were fabricated, and their compressive strength was evaluated in accordance with ASTM C39. Based on a comparative analysis of compressive strength under various curing conditions and alkaline activator compositions, the geopolymer waste form prepared using a 10 M NaOH solution with a sodium silicate–to–sodium hydroxide mass ratio of 1.5 exhibited the maximum compressive strength of 14.88 MPa.

The proposed solidification approach enabled a minimum waste loading of 70 wt%, exceeding the typical limit of approximately 30 wt% reported for conventional cement-based waste forms. This increased waste loading is expected to reduce the number of waste drums required for disposal, indicating that the proposed method provides a viable alternative for the treatment of spent zeolite waste while satisfying the mechanical strength requirements for disposal.

Keywords: *Solidification, Radioactive waste, Waste form, Zeolite waste*

S7-135

Construction Progress and Technical Overview of the Vrbina LILW Disposal Facility near Krško, Slovenia

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The low- and intermediate-level radioactive waste (LILW) disposal facility in Slovenia, represents a nationally significant infrastructure project for the permanent management of operational and future decommissioning waste from the Krško Nuclear Power Plant (NEK), as well as institutional radioactive waste generated in Slovenia. The facility is being developed to meet national and international safety requirements, ensuring long-term protection of people and the environment through a passive safety concept based on engineered and natural barriers.

The Vrbina facility is designed as a near-surface repository with a deep vertical silo structure. The disposal silo will have an internal diameter of app. 27 m and a depth of about 56 m, allowing emplacement of up to 990 reinforced-concrete disposal containers. Once complete, the silo will be sealed with a concrete cap and covered by a clay layer, forming a multi-barrier system consistent with international best practice for LILW disposal.

Construction of the repository is organised into three main phases. Phase 1, started in August 2023, delivered site infrastructure and physical protection of the site. Phase 2, focuses on civil works for the silo, and surface buildings. The diaphragm wall panels extend about 66 m deep to support the excavation and control groundwater. Phase 3 covers supply and installation of a portal crane and lifting equipment for safe handling and emplacement of waste containers. As of early 2026 construction is progressing according to plan. Preparatory groundworks and hydro-insulation were completed in 2024–2025, with excavation of the silo pit reaching design depth. Concrete placement for the silo floor slab has been executed and is now followed by continuation of silo concrete works, the most technically demanding part of the project. Surface facilities including the technological and administrative buildings are under construction, and trial operation of the repository is targeted for 2027–2028.

The disposal concept utilises standardised reinforced-concrete containers designed to accommodate combinations of different waste forms. This project integrates comprehensive civil engineering with nuclear safety principles, providing Slovenia with a purpose-built long-term disposal solution that fulfils ethical and regulatory obligations for responsible radioactive waste management.

Keywords: *Low- and intermediate-level radioactive waste (LILW), Vrbina disposal facility, near-surface repository, radioactive waste management, multi-barrier system, repository construction, diaphragm wall, reinforced-concrete containers, nuclear facility in Slovenia*

S7-137

NORM Disposal Sites at the Closed Uranium Mine Žirovski Vrh

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The closed Žirovski vrh uranium mine is in Slovenia, in the Poljane valley south of the village Gorenja vas. The uranium mineralization is hosted within middle Permian clastics of the Gröden formation, and the deposit was classified as medium-sized based on its reserves. Uranium concentrate production was carried out from 1984 until June 1990, when operations were terminated by government decree. The Act on the Permanent Cessation of Uranium Ore Exploitation (1992) defined the legal framework for mine closure, facility decommissioning, and long-term environmental protection. The Jazbec and Boršt disposal sites form the core infrastructure for permanent decommissioning and long-term containment of mining and processing (milling) residues. Remediation combined slope stabilization, multi-layer engineered cover systems, and hydraulic control measures to limit radon exhalation and prevent radionuclide and heavy metal migration into surface and groundwater. The Boršt disposal facility is located on the northeastern slope of Boršt Hill (535–565 m a.s.l.), covers 4,2 ha, and contains 610.000 tonnes of hydrometallurgical tailings, 111.000 tonnes of mine waste rock, and 9.450 tonnes of radioactively contaminated material, with a total activity of 48,8 TBq. It is sealed by a 2,05 m multi-layer cover. Slope instability was identified due to a reactivated paleolandslide; as a consequence, stabilisation works included construction of a rock-fill toe wall, a drainage tunnel and an array of sub-horizontal drainage boreholes. Residual movements are monitored by geodetic and GPS methods. The Jazbec disposal site lies in the Jazbec stream valley at elevations between 427.5 and 509 m a.s.l and contains 1,91 million tonnes of material, including 48.000 tonnes of Th-230 enriched red precipitate. The disposal body covers 6,6 ha, within a restricted area of 7,4 ha, with engineered slopes not exceeding 20° and a protective multi-layer cover of 1.95 m to limit erosion and water ingress. Following completion of remediation, both disposal facilities were designated as national infrastructure for long-term containment. The Agency for Radioactive Waste Management (ARAO) is responsible for their long-term stewardship, maintenance, and surveillance. Systematic monitoring ensures that effective dose contributions to the critical group do not exceed the regulatory limit of 0.3 mSv/year, and so far, results indicate that exposures in the vicinity of the sites remain below this threshold.

Keywords: *NORM Disposal sites, Žirovski vrh uranium mine, Monitoring, Agency for Radioactive Waste Management (ARAO)*

S7-163

Legacy Radioactive Waste Management and Development of a Radioactive Waste Processing Facility at the Vinča Site

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The management of legacy radioactive waste at the Vinča site is performed under the responsibility of the PC "Nuclear Facilities of Serbia" (PC NFS), which, upon its establishment, was assigned responsibility for addressing legacy radioactive waste and for the implementation of the objectives of the VIND Programme. Legacy radioactive waste stored in the closed hangars H1 and H2 consists of waste generated during the operation of the Vinča Institute, including the nuclear reactor facility and research laboratories in which sources of ionizing radiation were used or produced. The waste inventory also includes institutional radioactive waste originating from the former SFRY, generated by hospitals, oncology and biology institutes, radiochemistry laboratories, and industrial applications, including radioactive lightning rods.

The removal and processing of legacy radioactive waste and disused sealed radioactive sources (DSRS) from hangars H1 and H2 are being carried out in the Radioactive Waste Processing Facility (WPF). Following processing, radioactive waste is transferred to the new storage facility (hangar H3) in compliance with established waste acceptance criteria (WAC). Conditioned DSRS are transferred to a dedicated secure storage (SS) designed for high-activity sources, following treatment in the WPF and, where required, in the hot cell of the RA reactor.

A licence for the trial operation of the Radioactive Waste Processing Facility was issued by the Regulatory Authority in 2022. The first phase of the trial operation, conducted without radioactive and nuclear materials (cold phase), was initiated in 2022. The second phase of the trial operation, involving radioactive and nuclear materials (hot phase), was initiated in early 2025 and was successfully completed in the same year. Following the approval of the trial operation, documentation for the operational nuclear licence was submitted in December 2025. In parallel, licensing documentation for the decommissioning of legacy hangars H1 and H2 was submitted to the Regulatory Authority.

Decommissioning activities are planned on the basis of the existing waste inventory and the current condition of the legacy storage facilities. Initial activities comprise radiological surveys and characterization of hangar H2 and its surroundings, performed without waste manipulation in order to establish safe working conditions. These activities are followed by the removal and conditioning of DSRS, the removal of waste packages and bulky items for treatment in the WPF, and their transfer to interim storage in hangar H3. Owing to its degraded structural condition, hangar H1 is planned for immediate decommissioning through systematic characterization, decontamination and dismantling, including the removal of all systems and equipment. The continuation of legacy waste management and decommissioning activities is planned for 2026, subject to the issuance of the relevant regulatory licences.

Keywords: *legacy radioactive waste, radioactive waste processing, decommissioning*

S7-171

Information System Modernization for Safe and Traceable Management of Radioactive Waste at ARAO

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In 2024, ARAO initiated a comprehensive modernization of the information system supporting the management of radioactive waste. The legacy application, originally developed in 2007 to support waste acceptance, treatment, conditioning and storage at the Central Storage Facility (CSF), had become technologically outdated and insufficient to support new operational and regulatory requirements, particularly those associated with the forthcoming operation of the Low- and Intermediate-Level Radioactive Waste Disposal Facility. The project was procured as a turnkey solution through a public tender procedure and is being implemented in clearly defined phases, covering system design, redevelopment of the CSF module, migration of historical operational data, and development of a new module dedicated to disposal operations.

The new application preserves the existing waste management workflows while introducing enhanced traceability, structured data validation, role-based access control, and comprehensive audit trails, in line with nuclear safety and information-security requirements. Attention is given to maintaining continuity of regulatory reporting, documentation generation, and long-term inventory tracking across the transition from storage to disposal. The functional design explicitly integrates the operational logic of waste acceptance, package verification, inventory management, and administrative control, while establishing a digital framework for the future disposal process at Low- and Intermediate-Level Radioactive Waste Disposal Facility.

A key driver of the modernization was the transition from a storage-only system supporting a single facility to a lifecycle-oriented system integrating storage and disposal operations and accommodating both institutional radioactive waste and waste from the national nuclear power plant.

The paper presents the project objectives, procurement and implementation approach, and early operational experience. Special emphasis is placed on the migration of historical waste data from the legacy system, which proved to be less complex than initially anticipated, and on practical lessons learned when introducing a new digital system into a safety-critical radioactive-waste management environment. The project represents a key step towards ensuring long-term data integrity, operational transparency and regulatory compliance throughout the full lifecycle of radioactive waste management in Slovenia.

Keywords: *radioactive waste management; waste traceability; regulatory compliance; data migration*

S7-174

RD&D Program for joint SNF and HLW Disposal in Deep Geological Repository in Republic of Croatia or in Republic of Slovenia

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The Research, Development, and Demonstration (RD&D) Program for joint Spent Nuclear Fuel (SNF) and High-Level Radioactive Waste (HLW) Disposal in Deep Geological Repository (DGR) in the Republic of Croatia (RC) or in the Republic of Slovenia (RS) was prepared in the frame of the Fourth revision of the Krško NPP Radioactive Waste and Spent Nuclear Fuel Disposal Program. The RD&D Program was commissioned by the Croatian Fund for financing the decommissioning of the Krško NPP and the disposal of radioactive waste and SNF, and the Slovenian Agency for Radioactive Waste Management (ARAO), in accordance with national regulations and strategic documents of both countries. The Program was developed by an international consortium of Croatian, Slovenian and Czech research institutions.

Based on the Intergovernmental Agreement between the Republic of Croatia and the Republic of Slovenia, both countries are obliged to ensure the safe long-term disposal of their respective shares of SNF and HLW generated by the operation and decommissioning of the Krško NPP. National strategies envisage disposal of SNF and HLW in a deep geological repository (DGR) located either in Croatia or Slovenia, with an alternative option of an international repository if available.

The RD&D Program provides a comprehensive framework for implementing this objective, including a review of previous research, development of site selection criteria, proposal of technologies and methodologies, a phased timeline, pre-operational monitoring, stakeholder engagement strategies, and cost estimates for program implementation. The site selection process will be multidisciplinary, combining desk studies, laboratory and field investigations, as well as economic and sociological studies and analyses.

Keywords: *RD&D, DGR, SNF, Site selection criteria*

S7-178

Research, Development and Establishment Program for a Radioactive Waste Disposal Facility in the Republic of Croatia

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The Research, Development and Establishment Program for a Radioactive Waste Disposal Facility in the Republic of Croatia was commissioned by the Fund for financing the decommissioning of the Krško Nuclear Power Plant and the disposal of radioactive waste and spent nuclear fuel, in accordance with the Bilateral Agreement between Croatia and Slovenia, as well as national strategic and regulatory requirements. The Program was developed by a consortium comprising the Croatian Geological Survey and the University of Zagreb, Faculty of Mining, Geology and Petroleum Engineering.

The Program is based on the disposal of low and intermediate level radioactive waste (LILW) in a near-surface, vault-type disposal facility, combined with the possible disposal of institutional radioactive waste (IRW) in shallow boreholes at the same site. The Program defines disposal concepts, the implementation of research activities in line with the proposed site selection criteria across the Program's implementation phases, and a comprehensive timeline, supported by a simplified stakeholder engagement program and cost estimates.

The implementation of the Program covers all phases of disposal facility development, including research and development, site selection and design, construction, operation, closure, and is planned for the period 2026–2044. The Program provides a structured and realistic framework for the safe and long-term disposal of radioactive waste in the Republic of Croatia.

Keywords: *LILW, IRW, Disposal facility, Implementation phases*

S7-180

Results from Waste ERASER: Testing the Complete Treatment Process from Chemical Decomposition to Final Conditioned Waste Package

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Solidification and in particular cementation of spent ion exchange (IEX) resins often faces challenges, such as a low incorporation rate into the matrix, beads swelling, presence of combustible organic content and matrix inhomogeneity. The innovative process of spent IEX resin liquefaction avoids the common problems associated with IEX resin cementation. The liquefaction process reduces the organic content and results in a homogeneous distribution of radioactivity in the final waste form. The higher level of possible incorporation of treated IEX in the conditioned waste, e.g. cement matrix, offers significant cost-saving potential. In some cases, reducing the organic content of spent IEX resins is a requirement for disposal at repositories.

To address this issue, Framatome developed the Waste ERASER (Extended Volume Reduction by Advanced Treatment of Spent Ion Exchange Resins – formerly SPRINT) process for treating spent IEX resin including different possibilities of final conditioning. Initial on-site demonstration trials were conducted in 2020 at a nuclear power plant (NPP), involving the liquefaction of radioactive spent IEX resins from operation and cementation of the resulting liquefied resins. The produced specimens were then evaluated for leaching behaviour and compressive strength according to the Swiss waste acceptance criteria.

Following successful demonstration trials, Framatome designed, manufactured, and commissioned a semi-industrial scale pilot plant at Framatome's site in Karlstein, Germany. This pilot plant incorporated experience feedback gained from the on-site NPP trials and demonstrates how a flexible and simple liquefaction plant could be implemented in a NPP using a modular design.

Framatome has initiated a research and development project, partially funded by the German Ministry of Education and Research, utilizing its semi-industrial scale pilot plant. This R&D project is divided into two main phases. The first phase successfully focused on gaining operational experience with the process, emphasizing parameters for safe and efficient operation. The second phase focused on the immobilization of the resulting liquid concentrates using different matrices and recipes.

After the completion of both phases in 2025, the results will be presented. Different IEX resins and several simulated ion loads on the resins have been tested. The simulated ion loads have been selected to represent IEX resins used for waste water treatment during operation and also IEX resins which are used during a full system decontamination.

The liquefied IEX resins derived from the pilot plant were immobilized using different cement formulations, geopolymer formulations and drying techniques. Emphasis was placed on ensuring volume optimization and alignment with international waste acceptance criteria.

Keywords: *spent resin, immobilization, cementation, geopolymerization, volume optimization*

S7-195

NEK's LILRW Management and Handover Preparations for 2028–2030

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Low and intermediate-level radioactive waste (LILRW) represents an important segment of radioactive material management. It originates from the operation of nuclear power plants and research reactors, as well as from medical, industrial, and research activities. Although its radioactivity is lower than that of high-level radioactive waste, LILRW still requires careful planning for handling, storage, and final disposal to ensure the protection of people and the environment. Key challenges in LILRW management include ensuring the long-term stability of storage and disposal facilities, preventing contamination, and maintaining transparent communication with the public. The Krško Nuclear Power Plant (NEK), in cooperation with the Slovenian Agency for Radwaste Management (ARAO) and the Croatian Fund for financing the decommissioning of NEK and disposal of NEK radioactive waste and spent nuclear fuel (FOND), has been making great efforts in recent years to enable and implement the handover of LILRW to the final recipients. In accordance with the conclusions of the Coordination Committee and the Interstate Commission, Slovenia and Croatia are carrying out preparations for the LILRW transfer in the following areas:

- a new revision of the study on preparation for the transfer of LILRW to both transferees,
- the second revision of the study of the Possible Division and Takeover of the Operational and Decommissioning RW from Krško NPP,
- procurement of handling and transport equipment, and
- development of handover documentation.

Because the start of the handover process was postponed from 2023/2024 to 2028, NEK was required to provide additional temporary storage capacity for LILRW. Currently, the first four phases of the LILRW handover are planned for the years 2028, 2029, and 2030. The paper presents the preparation measures NEK has already implemented and evaluates the key challenges (technical, regulatory, and logistical) anticipated during the upcoming handover.

Keywords: *Low and intermediate-level radioactive waste (LILRW), NEK, LILRW Handover, ARAO, Fund NEK*

S7-203

Safety Assessment for the Centre for Radioactive Waste Management at the Čerkezovac Site in Support of Location Permit Licensing

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A Safety Assessment has been developed to support the establishment of the Centre for Radioactive Waste Management at the Čerkezovac site in the Republic of Croatia. The Centre is intended for long-term storage of half of the operational radioactive waste generated by the Krško Nuclear Power Plant (NPP), as well as institutional radioactive waste arising from medical, research, and industrial activities in Croatia. The planned facility comprises two storage units: a central storage facility for institutional radioactive waste (IRW) and a long-term storage facility for low- and intermediate-level radioactive waste (LILW) from the Krško NPP. The Safety Assessment was performed in support of the location permit licensing phase and is based on the preliminary design of the facilities. The assessment systematically evaluates safety aspects associated with radioactive waste transport to the site and the operational phase of both storage facilities. As an important component of the licensing process, the assessment provides a structured and comprehensive evaluation demonstrating that the proposed design and safety related systems meet relevant regulatory requirements and that associated risks are well identified, analysed, and reduced by proposed measures to levels that are as low as reasonably practicable. The paper describes the scope, structure, and methodological framework of the Safety Assessment, including the identification of initiating events, analysis of normal operation and postulated accident scenarios, and evaluation of radiological consequences. Dose assessments for workers and members of the public are conducted for normal operation, postulated accident conditions, and radioactive waste transport to the Centre. The results provide technical justification for licensing and confirm that the proposed design and operational conditions enable safe long-term storage of radioactive waste at the Čerkezovac site.

Keywords: *Safety assessment; Radioactive waste management; LILW; Institutional radioactive waste; Long-term storage; Dose assessment*

S7-233

Design and Implementation of the Defence-in-Depth Concept for a Radioactive Waste Management Centre in Croatia

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This paper presents the design and technical concept of a national radioactive waste management facility in the Republic of Croatia, addressing the safe and sustainable management of institutional radioactive waste, disused sealed sources, and the Croatian share of low- and intermediate-level radioactive waste generated by the operation and decommissioning of the Krško Nuclear Power Plant. The proposed solution is based on the establishment of a centralized waste management centre at the Čerkezovac site, comprising a Central Storage Facility for institutional radioactive waste and disused sealed sources, based on the adaptation of existing structures, and a newly designed Long-Term Storage Facility for low- and intermediate-level radioactive waste. The long-term storage is designed with a capacity of approximately 2,450 reinforced concrete containers arranged in multiple layers, enabling efficient use of space and operational flexibility. The facility is conceived as a passive storage system without on-site waste processing, where only conditioned waste packages are accepted, inspected, and stored. Operational activities include transport, receipt, verification, and periodic inspection in accordance with national, EU, and international regulatory requirements. A key feature of the design is the implementation of the defence-in-depth safety concept, based on multiple independent engineering barriers preventing radionuclide release. These include the waste matrix, reinforced concrete containers, structural confinement, and systems for controlled collection of potentially contaminated liquids. The concept is further supported by passive ventilation, radiological monitoring, and environmental surveillance. The proposed solution ensures a high level of radiological protection for workers, the public, and the environment, while meeting regulatory and strategic requirements. This work demonstrates a practical engineering implementation of the defence-in-depth concept in a predominantly passive storage system, providing a transferable model for similar facilities under regulatory and site-specific constraints.

Keywords: *Radioactive waste management, Low- and intermediate-level waste, Defence-in-depth, Passive safety systems, Engineering barriers*

Session 8
Small Modular Reactors (SMRs)

S8-108

Consideration of GWR Technique Adaptation for Pressurizer Level Measurement in i-SMR

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This study explores the application of Guided Wave Radar (GWR) technology for measuring the pressurizer level in the Innovative Small Modular Reactor (i-SMR), aimed at maximizing safety and operational efficiency.

Unlike conventional large-scale nuclear power plants, the i-SMR features an integral reactor structure that necessitates minimizing lateral penetrating piping. In turn, the GWR method, which allows for minimally invasive measurement through the top flange, is considered a promising alternative. GWR offers superior accuracy by directly measuring the liquid surface, overcoming the limitations of differential pressure-type gauges sensitive to coolant density changes. Furthermore, it enables continuous measurement—unlike Heated Junction Thermocouples (HJTC)—and is less susceptible to signal interference compared to ultrasonic methods. The primary design focus is ensuring survivability under extreme conditions. To maintain hermetic sealing under high temperatures (300°C) and high pressures (200 bar), the integrity of the probe module and internal sealing is critical. Verification will be conducted through empirical testing using a dedicated high-temperature, high-pressure chamber. Additionally, since the transmitter is located far from high-radiation areas, specialized Mineral Insulated (MI) cables, advanced connector technologies, and high-precision signal processing algorithms are required to detect microvolt-level echo signals despite signal attenuation and loss. Key performance requirements include an accuracy and repeatability within 10mm over an 8m measurement range, with a response time of less than 2 seconds. To compensate for level errors caused by dielectric constant fluctuations in the two-phase (water-steam) environment, a Time Domain Reflectometry (TDR) based reference measurement technique is applied. Lastly, for ease of maintenance, a weight is attached to the bottom of the probe, and a funnel-shaped supporter is designed on the surge plate to facilitate installation. A self-diagnostic function is also integrated to monitor sensor anomalies. The design requirements for GWR derived from this study are expected to serve as core technologies to enhance the reliability of the integrated instrumentation and control system of the i-SMR.

Keywords: *i-SMR, GWR, TDR, Pressurizer Level, Survivability*

S8-113

Physics-Informed Digital Twin and Verified Imaging Pipeline for Under-Lead Ultrasonics in Lead-Cooled SMRs

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Reliable operation of lead-cooled small modular reactors (SMRs) hinges on in-service diagnostics that work in optically opaque heavy liquid metal. Under-lead ultrasonics is one of the few viable modalities, yet performance is bounded by coupled physics and engineering constraints: high-temperature molten Pb propagation over long paths, and wetting-dependent coupling at steel/lead interfaces that destabilizes amplitude and phase. Instrument-rendered B-scans are therefore useful as a visual reference but are not physically linear ground truth for algorithm development.

We present a joint industry-university contribution that closes the loop from physics assumptions to verifiable software artifacts: (i) a parameterized digital twin and scenario library aligned to lead-cooled SMR constraints, and (ii) a versioned DSP/reconstruction pipeline with regression-testable metrics. The reference configuration is defined around a compact two-probe pitch-catch arrangement (77 mm spacing) with controlled target positioning (carousel concept) and operating conditions representative of under-lead environments: temperatures around 380 °C, controlled gas atmosphere for oxygen management (e.g., Ar + 1% H2 with scrubber and O2 monitoring), and stabilized wetting via thin Ni coatings on steel interfaces (approx. 1-3 μm). Test scenarios are organized as a five-case matrix (T1-T5) spanning wetting-limited noise baselines, attenuation/SNR reference conditions, and imaging cases with simple targets in the 7-10 mm range, including propagation distances comparable to reactor geometries (order 0.5-1 m). While the conceptual test setup anchors realistic boundary conditions, the current proof-of-concept phase is explicitly simulation-first; experimental qualification is treated as a follow-on step.

Hard-physics modelling uses a flow/gradient-aware propagation framework based on the Pridmore-Brown equation to bound when advection and shear terms can be treated perturbatively. For representative coolant velocities u ≈ 1 m/s and Pb sound speed c ≈ 1750 m/s, the Mach number is M ≈ 5.7×10^-4; we further use the oscillatory condition K ≤ 0.1 (ratio of wavenumbers) as a practical check when sweeping parameter sets. Coupling uncertainty is represented via an interface transmission factor to capture the wetting state and its impact on effective bandwidth and SNR.

On the software side, we deliver an A-scan core API with strict metadata handling and JSON I/O for reproducibility from RF waveforms to imaging inputs. In a reference regression test (synthetic steel A-scan, $c = 5900$ m/s; 10 mm indication at TOF ≈ 3.39 µs; 30 mm backwall at TOF ≈ 10.17 µs), the pipeline achieves $|\Delta t| \leq 0.05$ µs and $|\Delta \text{depth}| \leq 0.5$ mm and reports gate-resolved SNR metrics (~28-33 dB; global ~32 dB). Computational feasibility for large parameter sweeps is demonstrated via k-Wave acceleration in our benchmark (~96 h CPU vs ~2 h GPU). We will also report k-Wave datasets mapped to T1-T5 and reconstructed B/C-scan outputs (DAS/TFM, with optional plane-wave compounding) with sensitivity analyses over attenuation and coupling proxies, together with throughput and reproducibility KPIs. This provides a traceable evidence chain from inspection physics to maintainable code, supporting operation/maintenance and deterministic safety analyses for lead-cooled SMRs.

Keywords: *lead-cooled SMR; under-lead ultrasonics; wetting; Pridmore-Brown; k-Wave; A-scan; DSP; TFM; reproducibility*

S8-117

Development Strategies for Advanced Nuclear Instrumentation Technologies for Innovative Small Modular Reactors

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This paper presents development strategies for advanced nuclear instrumentation technologies for application in Korean innovative Small Modular Reactors (SMRs). Innovative SMRs are emerging as next-generation nuclear reactors with high potential in terms of safety, economics, and deployment flexibility. They adopt an integrated design in which major primary components, including steam generators, reactor coolant pumps, and a pressurizer, are installed inside the reactor pressure vessel, while a containment vessel (CNV) surrounds the reactor vessel to form a double containment structure. These design features impose severe environmental and structural constraints, such as high temperature, high radiation, limited installation space, and restricted accessibility, which limit the applicability of conventional instrumentation systems developed for large commercial nuclear power plants. To address these challenges, innovative SMRs require highly reliable and high-precision instrumentation technologies capable of accurately monitoring process variables and equipment conditions, as well as supporting a high level of automation and autonomous operation. This paper proposes a development strategy for multipurpose high-precision instrumentation technologies aimed at enhancing the safety and operational efficiency of innovative SMRs. An integrated instrumentation concept is presented, incorporating fibre-optic distributed sensors, low-power wireless multi-sensing technologies, process instrumentation sensors, ex-core neutron flux monitoring systems, and plant condition monitoring instrumentation tailored for innovative SMRs.

The results of this study are expected to support autonomous operation, equipment fault diagnosis, and predictive maintenance in innovative SMRs, and to serve as a key enabling technology for advanced instrumentation systems in innovative SMRs and future nuclear reactors.

Keywords: *Small Modular Reactor, Advanced nuclear instrumentation, Fibre-optic distributed sensors, RCS flow measurement, GWR-based level measuring, ex-core neutron flux monitoring*

S8-125

Individual Phase Flow Rate Measurement Technique Using Optical Fibre Sensor in Horizontal Stratified Flow

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Traditional flow meters, such as orifice and electromagnetic meters, are effective for total flow measurement but have limitations in precisely identifying the phase distribution across a pipe's cross-section in two-phase flows like stratified flow. Consequently, this study proposes a technique using a distributed optical fibre sensor to identify phase distribution and measure individual phase flow rates. The proposed method identifies the air-water interface and flow velocity through temperature profiles obtained from the optical fibre sensor, utilizing this information for individual flow rate calculation. Specifically, the interface is identified based on temperature characteristic changes resulting from heat capacity differences between water and air, while the velocity is calculated via differences in temperature profiles caused by variations in convective heat transfer. To validate this, a horizontal pipe experimental setup simulating water-air stratified flow was developed. A sensor with a spatial resolution of 0.65 mm and a sampling rate of 15.61 Hz was inserted vertically, securing temperature data from 148 measurement points. Experiments were conducted with water flow rates up to 13 L/min and air flow rates up to 55 L/min to cover laminar, transitional, and turbulent regimes. The analysis results showed that the interface heigh could be successfully identified through distinct temperature gradient changes at the air-water interface. Furthermore, it was confirmed that the maximum temperature difference in a quasi-equilibrium state varies due to differences in the convective heat transfer rate corresponding to velocity changes, validating its potential as a velocity identification index. This study experimentally verified the proposed measurement concept and suggests practical applications for individual phase flow measurement in stratified flows.

Keywords: *Distributed temperature sensing, Rayleigh backscattering, Interface measurement, Flow velocity measurement, Air-water stratified flow*

S8-140

Uncertainty Analysis of Wall-Temperature Measurement using Distributed Optical Fibre Sensors

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Monitoring thermal transients at the tube wall is essential for evaluating the heat transfer performance and structural integrity of steam generators in Small Modular Reactors. Distributed Optical Fibre Sensors based on Rayleigh backscattering offer a significant advantage over conventional point-type sensors due to their high spatial resolution, which enables the detection of localized temperature fluctuations. However, the measurement reliability of these sensors during rapid thermal changes must be quantitatively verified. This study performs an uncertainty analysis of optical fibre sensors for transient wall-temperature measurements conducted in a test section designed to observe thermal-hydraulic behaviour. The experimental procedure used a valve-switching method to induce rapid thermal transients at the test section. To evaluate the reliability of the measured data, the uncertainty was determined by classifying errors into bias and precision components. In particular, the bias error analysis accounts for the physical installation of the stainless steel capillary tubes containing the optical fibre sensors, which were mounted along the wall of the flow path. This includes potential errors arising from the spatial positioning of the STS tubes and the thermal contact resistance between the sensor, the tube, and the wall surface. By combining these systematic factors with data acquisition limits and statistical fluctuations, the total measurement uncertainty was successfully evaluated. The analysis focuses on how various error factors affect temperature measurements during rapid thermal changes and fast-shifting boundary conditions. By integrating the experimental setup characteristics with signal processing factors, this approach provides a systematic framework for estimating uncertainty when temperature differences change quickly over time. The methodology presented in this study is intended to establish a clear procedure for verifying thermal measurements, which can be applied to improve the data reliability of thermal-hydraulic monitoring systems.

Keywords: *Uncertainty Analysis, Optical Fibre Sensors, Random Error, Bias Error*

S8-166

Advanced Modular Reactor Development in the Slovak Republic

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Newvys is a joint venture of newcleo and JAVYS companies established in June 2025, with a mission to deploy Generation IV reactors at Jaslovské Bohunice in Slovakia on the former site of Nuclear Power Plant V1. The innovative Advanced Modular Lead Fast Reactor (LFR-AS-200) technology, cooled by liquid lead and operating on the basis of fast neutrons, will enable the reuse and reprocessing of spent nuclear fuel. The technology has the potential to significantly reduce the volume of spent nuclear fuel and the radiotoxicity of radioactive waste. The exclusive use of spent nuclear fuel would also reduce Slovakia's dependence on imports of fresh fuel.

Keywords: *SMR, AMR, advanced, modular, spent fuel*

S8-192

Westinghouse Advanced Passive (AP) Technology

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Westinghouse's Advanced Passive (AP) reactor technology has undergone a structured, evidence based evolution over more than three decades, beginning with the AP600 and leading to the deployment of the AP1000 Pressurized Water Reactor (PWR), and remains at the heart of the AP300 Small Modular Reactor (SMR). This progression reflects a consistent design philosophy: leveraging natural forces—gravity, natural circulation, convection, and stored energy—to achieve safe shutdown, maintain core and containment cooling, and ensure long term heat removal without operator intervention or reliance on AC power. The AP600 established the foundational passive safety system architecture, validated through extensive separate and integral effects testing programs. These tests confirmed the capability of passive core cooling and containment systems to maintain safety margins under a wide variety of postulated accidents, establishing regulator confidence and enabling first-of-a-kind certification. Building upon AP600, the AP1000 introduced scale up efficiencies, simplified systems, improved constructability through modularization, and enhanced reliability via standardized components. With global regulatory approvals and operating experience exceeding multiple reactor years, the AP1000 PWR validated the robustness, reliability, and deployability of advanced passive nuclear technology. The AP300 SMR represents the latest product in the Westinghouse AP family, directly inheriting its core design, major equipment, proven fuel, and safety systems from the AP1000 PWR, while scaling down to a 300–330 MWe class reactor optimized for deployment flexibility. Its passive safety systems maintain the same fail safe, self sufficient characteristics: automatic actuation without pumps, diesels, or operator actions; long duration passive cooling; and a containment architecture protecting all critical systems within a steel vessel and shield building. Testing needs for AP300 are largely met by the AP600/AP1000 data sets, significantly reducing FOAK risk. This paper will summarize the development and maturation of the Westinghouse passive safety systems and their integration into the AP1000 PWR and AP300 SMR designs.

Keywords: *AP300, design architecture, SMR, validation, Westinghouse*

S8-202

A Coolant-Free Generic Replacement for a Reactor Cavity Cooling System: Steady-State Normal Operation Analysis of Implementation for a Sodium-Cooled Fast Reactor

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A new device that replaces active Reactor Cavity Cooling Systems (RCCS) is proposed. The device acts as mostly an insulator during normal operations as the bulk of in-core generated heat is evacuated via the normal reactor cooling path into the plant's energy conversion system. When the reactor is shutdown or if normal cooling paths are not available, the device transitions from a heat path into an ultimate heat sink. The device does not use cooling fluids and instead performs its functions via conduction and radiative heat transfer. Passive RCCS designs often rely on continued availability of a coolant fluid and a path for the fluid to ensure natural convection heat transfer to an outside sink. The new device relaxes the need for fluids and thus makes reactor deployment site-neutral (e.g., possible in arid regions). Also, the device does not require the availability of a coolant path. Yet it is passive, needing no external energy supply and no operator intervention.

The Passive Reactor Radiative Thermal Sink (PaRRaTS) system relies on a radiative heat transfer "valve" made of two shells to control the heat flow out of the reactor vessel wall into a surrounding soil heat sink. The inner side of the outer shell of PaRRaTS is fitted with metal fins that extend into the surrounding soil to deposit removed heat. The inner shell is also fitted with fins that interweave between those of the outer shell without contact. Thus, at lower temperatures PaRRaTS is essentially insulating, while at higher inner temperatures net radiative heat transfer from the inner to the outer shell becomes significant. The system is modeled using Idaho National Laboratory site soil and its performance is assessed. Annual fluctuations in soil makeup and temperatures are considered in setting boundary conditions. Another model input includes the variety of material properties for each section of the PaRRaTS device. The materials are selected based on their function within PaRRaTS. The selection optimizes thermal conductivity, specific heat, and emissivity with cost limits as a constraint. The effectiveness of this system is compared to the Reactor Vessel Air Cooling System (RVACS) of a model sodium-cooled fast reactor. The objective is a nearly insulating system under normal operating conditions, removing less heat than RVACS, that reverts to a more conducting one removing more heat following a SCRAM.

Performance is assessed by the maximum temperature of the soil and the radiative heat transfer between the two PaRRaTS shells. A steady state model is used for normal operating conditions, and steady state and initial transient conditions are used and compared for SCRAM. The ABAQUS code is used for the finite-element thermal analysis of PaRRaTS. A Python script facilitates the process. Results show that during normal operations PaRRaTS does not significantly heat up the soil, nor does it remove more heat than traditional RVACS. SCRAM conditions are addressed in a companion paper.

Keywords: *nuclear energy, heat transfer, passive safety, ultimate heat sink, reactor cavity cooling system, reactor vessel auxiliary cooling system, RCCS, RVACS, thermal valve, passive cooling, heat transfer*

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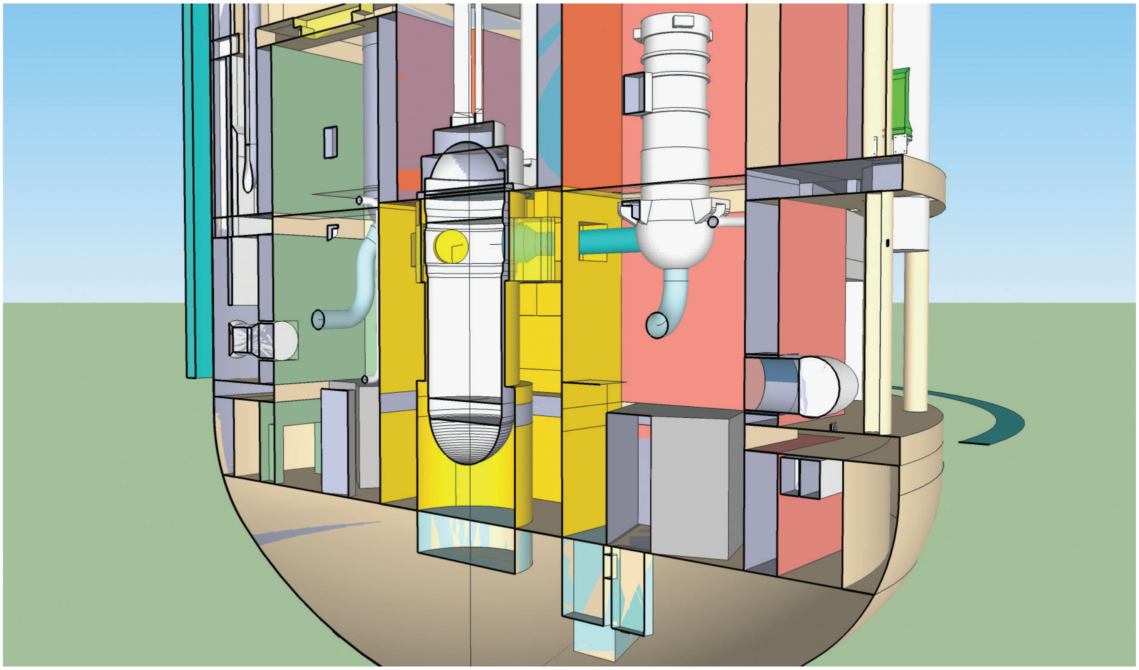

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