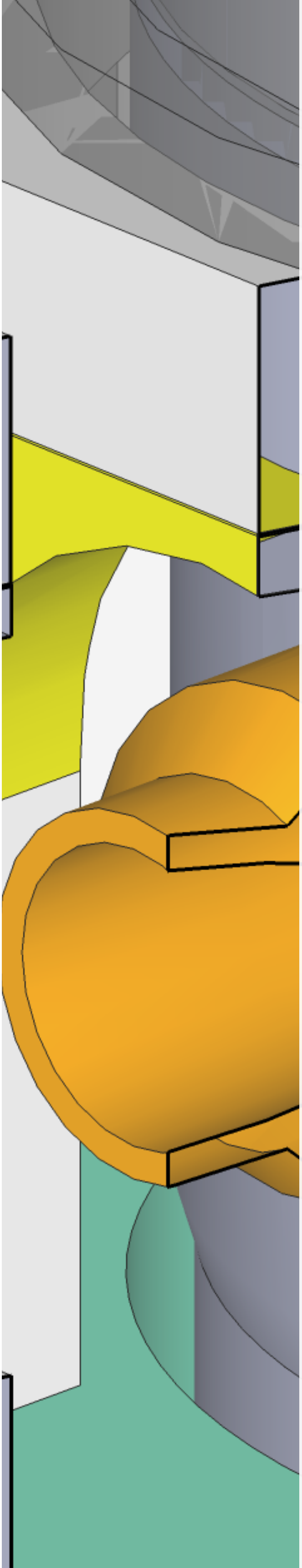

Croatian Nuclear Society

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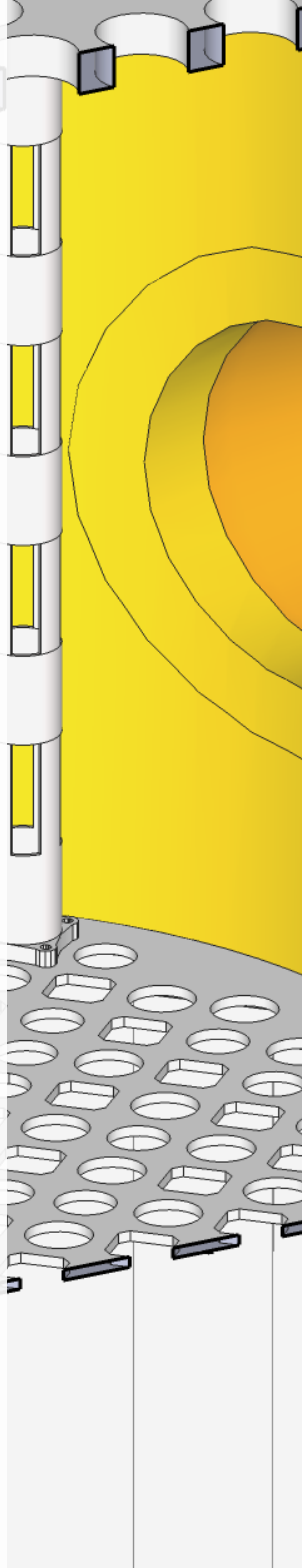
13th International Conference of the CROATIAN NUCLEAR SOCIETY

Nuclear Option for CO_2 Free Energy Generation

June 5-8, 2022, Zadar, Croatia

Book of Abstracts



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Foreword

This International Conference of the Croatian Nuclear Society, subtitled "Nuclear Option for CO2 Free Energy Generation", is already a 13th event in the successful series of international conferences organized biennially by the Croatian Nuclear Society, which was formerly known under the subtitle "Nuclear Option in Countries with Small and Medium Electricity Grids". This 13th event comes, after two postponements, as a first conference in the series since the outburst of the COVID pandemic. The numbers of participants and papers submitted clearly show that the interest in the subjects related to the nuclear option does not diminish.

The purpose of the conference series remains to be presenting and discussing 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 or medium electricity grids. The conference series was initiated in 1996 with the first conference taking place in Opatija. It was followed by seven conferences in Dubrovnik and the last four which were held in Zadar. This 13th conference, the last one in the series for now, takes its place in Zadar again.

In the societal context which is considerably marked by dynamic economic growth and concerns which include energy resources and their stability, potential climate changes and increasing greenhouse gas emission the issue of ensuring reliable and sustainable energy is becoming ever more challenging. Nuclear energy, as the CO2 free energy source, should have its place in resolving these concerns and enabling all the countries to cope with challenges under such context.

In view of good response and success of the previous conferences in the series, this 13th International Conference concentrates on these topics, as well as on those which attracted the most interest in previous conferences in the series. The Conference addresses the most relevant aspects of safety assessment of nuclear energy option. Additionally, it considers the nuclear option also from the point of view of the national energy strategies, resources, costs, technological, organizational and educational requirements, as well as environmental advantages. It, also, focuses on the matters related to the operation and design safety, fuel cycle, waste management, and decommissioning.

The important goal of the Conference, as in the previous cases, 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. The importance of international cooperation for the assessment of the nuclear option has been recognized by the International Atomic Energy Agency (IAEA). As a result of this recognition, the Conference is organized in cooperation with the IAEA. European Nuclear Society, and University of Zagreb, Faculty of Electrical Engineering and Computing took considerable part in the preparation and organization of the Conference.

Topics which are addressed by different sessions at the Conference reflect the above mentioned aspects of safety assessment, facility operation and design, fuel cycle, energy strategies at national level, regulatory practices, radioactive waste management, decommissioning, as well as many others. The papers presented at different sessions promote international exchange of experience and co-operation among the interested parties in these fields.

Authors' and presenters' contributions are provided in 8 invited presentations and lectures and 66 contributed papers.

The invited presentations address as diverse subjects as applications of small modular reactors, need for enthusiastic young nuclear professionals in order to ensure the future of nuclear energy, role of nuclear option in achieving net zero, design and safety assessment and verification of different reactor technologies (including the advanced reactors), status of the new NPP project in Slovenia and developments in the field of accident tolerant fuel materials.

The contributed papers are grouped into eight thematic sessions:

- S1: Nuclear Safety Analyses (NSA)
- S2: Operation and Maintenance Experience (OME)
- S3: Nuclear Energy: Planning, Environment and Technologies (NEPET)
- S4: Regulatory Practice and Emergency Preparedness (RPEP)
- S5: Reactor Physics and Nuclear Fuel Cycle (RPNFC)
- S6: Severe Accident Analyses and Risk Assessment (SAA)
- S7: Radioactive Waste Management and Decommissioning. Radiation Hazard and Protection. (RWMD)
- S8: Safety Culture, Knowledge Management and Public Relations (SCPR)

As with the previous conference in the series, a topic of particular interest remains the role of Small Modular Reactor (SMR) designs in nuclear energy program and possibilities of inclusion of SMRs in the long term energy strategies. This topic is a subject of one among the invited presentations and is additionally addressed under the panel discussion, which is one of the focal points of the conference.

The main topics of the panel discussion were, for this conference, selected to be some basic preconditions needed for the nuclear option to become a relevant part of the solution of the global climate problems. For this purpose, the panel discussion is divided in two parts. The first part discussed the role and the importance of the nuclear option in national energy strategies while the second part is devoted to the challenges of the public relations for the nuclear option.

This Book of Abstracts provides printed abstracts of contributed papers, invited presentations and summarized outlines and topics for the panel discussion. The Proceedings with full papers will be provided for download from the Conferences website after the review process is finalized.

We would like to express our gratitude to not much less than 200 authors and co-authors that put a large effort into completing their 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 Conference.

Special acknowledgments are given to the International Atomic Agency and the European Nuclear Society for their provisions.

Last but not least, 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 2022

Editors

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Session 8 Safety Culture, Knowledge Management and Public Relations (SCPR)

Invited Presentations (IP)

IP-1

IAEA Support to Member States on Small Modular Reactors and their Applications

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Global interest in small modular reactors (SMRs) has been rapidly increasing due to their prospects to meet the needs of a wider range of users and applications, as well as to their expected contribution to the transition to net zero. The driving forces in the development of SMRs are their specific characteristics. They can be deployed incrementally to closely match increasing energy demand, resulting in a moderate financial commitment for countries or regions with smaller electricity grids. SMRs show the promise of significant cost reduction through modularization and factory construction which should further improve the construction schedule and reduce costs. SMR designs and sizes are also better suited for partial or dedicated use in non-electrical applications of nuclear power such as providing heat for industrial processes, hydrogen production or sea-water desalination. Process heat or cogeneration result in significantly improved thermal efficiencies leading to a better return on investment. SMRs also provide adaptability in terms of location of deployment since they may serve regions that are more difficult to support with other clean energy systems, like large nuclear power plants. These can be off-grid areas difficult to access, remote islands, or sparsely populated regions with small electric grids and limited infrastructure. This is the case of some nuclear embarking countries. Last but not least, the reduced nominal electric power, equal or lower than 300 MWe, along with the use of fuel different than the one in use in large NPP, ensure a source term lower of one order of magnitude and with a different isotopic composition than the source term associated to the release to the environment of radioactive materials, in case of accident, from the core of an existing large NPP. Therefore, Emergency Planning Zones (EPZs) and Emergency Planning Distances (EPDs) will likely be smaller for an SMR than for a large existing NPP.

Since several years the IAEA has been supporting interested Member States on a large spectrum of SMR related activities: from technology development and deployment to economic and financial considerations, from applicability of current safety standards and security guides to safeguard-by design and 3S, from dedicated fuel and associated fuel cycles to waste management and decommissioning, etc.. The IAEA has also started to apply the very well-known “milestone approach” and “reactor technology assessment” to embarking countries which have planned to include SMRs in their energy mix. However, Member States are asking for an Agency’s consistent and coordinated support related to all aspects of SMRs’ development, deployment, and oversight. The Standing Advisory Group for Nuclear Energy and the Commission of Safety Standards have stated that concerted and coordinated action by the entire

Agency is necessary to provide effective and efficient support to Member States and stakeholders interested in the early deployment of SMRs and their related electric and non-electric applications.

To respond to such requests, the IAEA engaged in a comprehensive and holistic effort to establish an Agency-wide Platform on SMRs and their applications. The Platform aims at supporting Member States in the early deployment of SMRs, including in accelerating their technology development, readiness level, and demonstration, analysing their competitiveness with respect to other clean energy technologies. At the same time, the Platform ensures that high standards of safety, security, and safeguards are considered at all stages.

The IAEA Platform on SMRs and their applications was established in April 2021 by the IAEA Director General with the purpose of coordinating the Agency's activities on SMRs and their applications and providing a "one-stop shop" for Member States and stakeholders. It is also linked to the brand new IAEA Nuclear Harmonization and Standardization Initiative (NHSI).

After shortly presenting the status of the SMR technology and its short- and medium-term prospects, this contribution describes some relevant ongoing IAEA activities on SMRs with a special focus on the tasks performed by the SMR Platform and the link with NHSI.

Keywords: *small modular reactor, non-electric applications of nuclear power, IAEA Platform on SMRs*

IP-2

The Future of Nuclear Power Depends on Enthusiastic Young Nuclear Professionals

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Nuclear fission (and fusion in the future) could provide a significant contribution to the global efforts to secure sufficient, environmentally acceptable and affordable energy. Unparalleled energy density, very low impact of the nuclear energy to the public health and environment, and the maturity and stability of the industry and regulators, are thoroughly supported by the available scientific and technical knowledge.

Now it might not come as surprise that the major challenge for the long-term success of the nuclear energy (and industry in particular) appears to be the dwindling public acceptance. We will discuss some possible causes for such developments in the past. We will also develop possible approaches that may enable the nuclear power to contribute much stronger to the future challenges of humanity. In particular, we will focus on developing and enabling enthusiastic future generation of nuclear professionals. This will require personnel with outstanding knowledge, skills and motivation. Apart from being technical specialists understanding the installations of increasing technical complexity, the new nuclear talents will have to be prepared to work and communicate in increasingly multidisciplinary, multicultural and highly competitive environments.

European Nuclear society (ENS) is fully aware of the sheer complexity of this challenge and calls for high level of support, coordination and partnership between all nuclear stakeholders, especially those involved in all levels of decision-making.

Keywords: *long term success of nuclear energy, knowledge, skills, public acceptance, education*

IP-3

NEXTRA Progress toward Design and Licensing of Molten Salt Research Reactor (MSRR)

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Molten salt reactors (MSRs) have several attractive features including operation at high temperature resulting in high thermal efficiency and suitability as industrial heat source, as well as operation at low, near-atmospheric pressure promoting enhanced safety. MSR feasibility was demonstrated with the Molten Salt Reactor Experiment (MSRE) that successfully operated at ORNL in 1964-1969.

Over the last two decades there has been renewed and ever-increasing interest in MSRs. A number of commercial and academic institutions in the US and worldwide are actively developing MSR designs. NEXT Research Alliance (NEXTRA) is a four-university consortium, including the Abilene Christian University (ACU) and its Nuclear Energy eXperimental Testing (NEXT) Lab, Georgia Institute of Technology, Texas A&M University, and The University of Texas at Austin. Supported by funding of Natura Resources LLC, NEXTRA has set out to design, license and commission a molten salt research reactor (MSRR) to be hosted on ACU campus.

This talk will present the progress toward and status of the design and licensing of MSRR, from the NEXT Lab formation at ACU (in 2016), to NEXTRA formation (formally initiated in 2019, completed in 2020), to the current design efforts, planned Preliminary Safety Analysis Report (PSAR) submittal to the US Nuclear Regulatory Commission (US NRC), and construction of the Science and Engineering Research Center (SERC) that will host MSRR.

Keywords: *Molten Salt Reactor (MSR), Molten Salt Research Reactor (MSRR), licensing of non-LWR reactors*

Note:
This invited talk reports on the work performed by many dedicated MSRR/NEXTRA/Natura team members.

IP-4

Advanced Reactors: Verification and Validation Needs

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The objective of the seminar is to highlight the key research initiatives and findings to support both the experiment and modeling needs for advanced reactor concepts. The presentation will provide a technical overview of the ongoing efforts in the areas of, the Versatile Test Reactor, focusing on the gas cooled cartridge loop development, Thermal Energy Distribution System under the Integrated Energy System Program, and the ongoing efforts to achieve accelerated development and deployment of microreactors, with a focus on design and development of a non-nuclear microreactor test bed to support modelling tool validation efforts. Access to experimental data from ongoing work will significantly increase the knowledge base for fundamental understanding and assist with uncertainty quantification.

Keywords: *advanced reactors, microreactors, Versatile Test Reactor, modelling and validation, non-nuclear test bed*

IP-5

Achieving Net Zero with Nuclear Energy

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Nuclear power is a resilient, firm, and flexible source of low carbon power that contributes to the security of energy supply and helps reduce dependence on fossil fuels. Over the past 50 years, nuclear power has cumulatively avoided the emission of about 70 gigatonnes (Gt) of carbon dioxide (CO2) and continues to avoid more than 1 Gt CO2 annually.

The use of nuclear power continues to grow, albeit more slowly than many other low carbon sources. Today, it provides 10% of the global electricity production. In 2020, the number of IAEA Member States operating nuclear power plants increased to 32. Most plan to expand their nuclear power capacity. Nearly 30 newcomer countries are embarking on, or considering, nuclear power. These are small but encouraging steps: nuclear generation will need to double by 2050 if we are to reach our net zero climate goals.

While electricity generation is responsible for close to 40% of the global CO2 emissions produced by the energy sector, the other 60% or so is generated primarily using fossil fuels in industry, heating in buildings and transport. Of all low carbon energy sources, nuclear power is one of the few, if not the only one, that can generate electricity, heat and hydrogen at scale. While crucial to reaching net zero, the potential of innovative nuclear systems such as Small Modular Reactors to produce these low carbon energy carriers is not sufficiently reflected in policies and investment decisions, which risks delaying their deployment.

As recalled by the recent reports of IPCC, building climate resilient infrastructure is of utmost importance to reduce future risk exposure and associated economic losses. Nuclear power plants are not immune to changing weather conditions. Heat waves, floods, severe winds etc cause occasional outages at nuclear power plants but data analysis from reports of weather induced power outages from IAEA Member States confirm the strong reliability of nuclear power plants in the face of extreme weather conditions. Regular adaptation upgrades of nuclear power plants contribute to the overall climate resilience of energy infrastructure.

Nuclear energy is also well suited to powering new economic development pathways of the fourth industrial revolution. New technologies, including advanced nuclear designs, are expected to enable emerging and developing economies to bypass (or 'leapfrog') conventional historical development paradigms built around fossil fuels and energy intensive industry, and instead put low carbon energy technologies — including renewables and nuclear — at the heart of economic development.

Reaching net zero without nuclear may be achievable, but it will be much more costly, and risky. Hence, reassessing the contribution of nuclear power to transitions towards resilient net zero energy systems is necessary.

Keywords: *energy transition, net zero, resilience, economic development*

IP-6

Validation, Uncertainty, Scaling, Passive Systems: AP-1000 Challenges

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The development of the AP-1000 design and of its precursor AP-600 started in the aftermath of the Chernobyl event (1986) when, among the other things, the need came out from scientific and technological community for a system resilient to deliberate threats by humans. Then, the 'passive-system' design-concept became relevant. The first AP-1000 entered in operation around three decades after that event. The issue in this presentation is how much the progress in nuclear science and technology, since the end of 1980's, affected the design of AP-1000. Five interconnected areas are identified, or (a) reliability of passive systems; (b) scaling and uncertainty; (c) coupling between three-dimensional neutron physics and thermal hydraulics; (d) consideration of large break loss of coolant; (e) simulation of instrumentation and control systems. All the areas are relevant for AP-1000 and standard Pressurized Water Reactors (PWR); however, the areas (a) and (b) have specific applicability for AP-1000 and constitute the main concern for the presentation. The conclusion from qualitative investigation is that safety demonstration did not take full benefits from progress in those areas: inadequacies characterize the scaling database and processes for determining the reliability of thermal hydraulic passive systems did not receive proper attention.

Keywords: *AP-1000 design; safety demonstration; reliability of passive systems; scaling and uncertainty*

IP-7

Status of the Project for the Construction of a New Nuclear Power Plant JEK

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The electricity generation sector is very important for Slovenia and the EU, so it is necessary to strive for a stable, efficient, and sustainable energy supply. At the end of 2020, the EU proposed raising the target for reducing greenhouse gas emissions by 2030 to at least 55% compared to 1990, which was also confirmed by the European Commission, and the EU plans to phase out fossil fuels by 2050.

The national strategy in Slovenia includes, in addition to efficient energy use, promoting the use of renewable sources, promoting cogeneration, and preventing climate change, guidelines on projects to achieve sustainable development of Slovenia such as extending the operational lifetime of NEK and building a new Krško Nuclear Power Plant (JEK2). Given the excellent experience in the use of nuclear energy, the choice and continuation of the nuclear option is a natural choice.

With the adoption of the National Energy and Climate Plan (NEPN) and the Climate Strategy in Slovenia, GEN gained a strategic basis for the continuation of the JEK2 project. The key steps that have taken place in the recent period are obtaining an Energy permit in July 2021, preparing and submitting documentation for the initiative to start the spatial planning to the Ministry for Infrastructure as the initiator and handing over the application of the initiator to the Ministry for environment at the end of March 2022. Several site investigation studies are in progress like Seismic and geology, Analysis of site selection variants, Analysis of the operation of cooling towers and their impact on the environment, in preparation for the environmental report within the Environmental Impact Assessment. Other major activities are related to the preparation of the requirements for suppliers to start negotiations (Request for Vendor Information), projection of staffing requirements and preparation for the establishment of JEK2 organization, participation in European Utility Requirement group (EUR) design assessments and SNSA licensing (nuclear licensing of JEK2 project).

Keywords: *JEK2, newbuild, NEK*

IP-8

Development and Assessment of an Extended MATPRO Materials Property Library for Accident Tolerant Fuel Materials

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The MATPRO materials property library, developed originally by the U.S. Nuclear Regulatory Commission in the 1970s for use in their steady state and transient fuel behavior codes, FRAP-CON and FRAP-T, and extended in the 1980s for SCDAP/RELAP5, has been widely used to describe the behavior of typical LWR core and RCS structural materials for both design and beyond design basis accident conditions. In the mid-1990s, this library was extended by Innovative Systems Software (ISS) to include the Zr-Nb cladding materials used in many VVER fuel assemblies, allowing the RELAP/SCDAPSIM and new integral code, ASYST to support the analysis of VVER-specific fuel bundle heating and melting experiments performed in the German CORA and QUENCH and Russian Parameter facilities for beyond design basis conditions. In 2021, it was proposed that this library be further extended to include a range of Accident Tolerant Fuel (ATF) materials to support an IAEA collaborative research project designed to extend the assessment and application of such materials. This activity is now well underway with support from ISS's collaborative partners from Mexico, Egypt, Spain, and Jordan as well as other collaborators in the IAEA project. This paper describes activities involved in this project, the development of material property models and reference correlations for one of the materials, FeCrAl, the implementation of these models and correlations into the SCDAPSIM package used in both RELAP/SCDAPSIM and ASYST, and the application of these new models and correlations in the analysis of a recent German QUENCH experiment, QUENCH-19 and representative PWR during the early phases of a SBO accident.

Keywords: *RELAP/SCDAPSIM, ASYST, PWR SBO analysis, Accident Tolerant Cladding, FeCrAl*

Note:

The authors have submitted a full paper for the Conference, which is the basis for the invited presentation. Presenting author is Ms Marina Perez-Ferragut. The invited presentation is broader than the paper itself as it also provides an introduction to Accident Tolerant Fuel and discussed its importance for the nuclear energy option.

Panel Discussion (PD)
Nuclear for Climate – Some Basic Preconditions

PD-1

Nuclear in National Energy Strategy

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Electricity generation is currently dominated by fossil fuels with coal and gas responsible for 63% of total production. These fossil fuels contribute about 40% to the world's CO_2 emissions.

Nuclear power is a low carbon energy source, with greenhouse gas emissions lower than most renewable sources, and comparable to wind sources.

Since 1970, nuclear power has prevented the emission of 64 Gt CO_2 equivalent. However in order to meet the Paris agreement of limiting rise in global temperatures to below 2 degrees, nuclear power capacity needs to increase.

The world is facing an energy crisis. Prices are soaring and many countries are striving to have net zero carbon. So how do we get clean energy to everyone sustainably?

A balanced green energy system that utilises low carbon technology is critical. Renewables like wind and solar are part of the solution but we also need nuclear. It's a proven technology which is safe, reliable and not weather dependent. It's also ready to rise to the challenge.

Supporters of nuclear energy say it can help us wean our economies off polluting fossil fuels. No surprise, it's a heated issue. But what about the facts? Can nuclear power really help save the climate?

In the four most important scenarios of the IPCC to limit global warning to 1.5 degrees Celsius, the use of nuclear energy increases.

Is nuclear power part of the climate solution?

Nuclear power can play an important role in clean energy transitions. A doubling in annual capacity additions is needed to be on track with the IEA's Net Zero Scenario

After a bitter political battle, the European Commission has opted to include nuclear in its sustainable taxonomy as transition activities. Supporters of nuclear power, including 12 EU member states who publicly backed its inclusion, say that nuclear is a low-carbon power source that must be part of any energy mix to tackle climate change, and does not cause more significant harm than other industries included in the taxonomy. They say that the science, and evidence-based policy support its inclusion. Opponents say that it should not be included because radioactive waste means it is not sustainable.

Apart from public opinion, one of the biggest barriers to nuclear energy is financing. It is needed to create better preconditions for nuclear energy, provide financial support where necessary and keep the nuclear power plant

Some questions:

- What might a balanced energy eco system look like, and what would be the role of nuclear?
- Small Modular Reactors: The next generation of nuclear power stations: making nuclear affordable and investable.
- Hydrogen: Synergies with nuclear that increase energy system resilience and efficiency.
- Is nuclear energy good for the climate?

Keywords: *nuclear power as a part of the climate solution; role of nuclear in a balanced energy eco system; financing of nuclear energy option; small modular reactors in nuclear option; hydrogen production by nuclear as synergy; national nuclear programs*

PD-2

Importance of Public Communication in Nuclear

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The Round Table will address the importance of public communication and public involvement in nuclear decision-making processes. The panellists are going to discuss the necessity of early and two-way communication, mutual trust and cooperation among stakeholders. Comments and opinions from audience will be welcomed and interaction on this issue will be encouraged.

Research and experience in some countries shows that sound communication and stakeholder involvement are essential components for conducting nuclear-related projects. Most European countries have established dialogue processes, which take different formats and involve a wide range of actors. Through these processes, it has been demonstrated that the involvement of potential host and neighbouring communities, regulatory bodies and experts is crucial. Transparency and openness are also pre-conditions for trust building.

There are national and international commitments to stakeholder involvement. In order to build trust, stakeholder involvement should take place in every stage of the life cycle of nuclear facilities and should occur regularly and frequently. Communicating with stakeholders appears on every level of IAEA safety standards: from safety fundamentals through requirements to guides. They are handled specifically, laying out the responsibilities of the governments, regulators, utilities and academia. There are also national and international commitments to involving stakeholders. Numerous challenges regarding public perception are encountered during these processes and overcoming them should be our mutual goal

Keywords: *public communication, public involvement, stakeholder involvement, public communication in nuclear, transparency in nuclear*

Session 1
Nuclear Safety Analyses (NSA)

S1-113

RELAP5/mod3.3 Analysis of Natural Circulation Cooldown with One Inactive Loop for Nuclear Power Plant Krško (NEK)

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If the Reactor Coolant Pumps (RCPs) in a pressurized water nuclear power plant are stopped then a loss of forced reactor coolant flow will occur and the decay heat from the core to reactor coolant and from reactor coolant to the steam generators (SG) will be removed by natural circulation. Ideally, all of the reactor coolant system (RCS) loops will be active and participate in the natural circulation cooling process. However, if certain failures occur, one loop may become inactive and that SG would not be available for cooling the RCS. If a natural circulation cooldown is initiated at too high rate using the active SG, the transfer of heat to the inactive loop SG will lag the conditions in the remainder of the RCS, such that the density driving head from the downcomer/core region portion is negated. As the RCS flow in the inactive loop slows down, it can eventually stop or stagnate as a result of this excessive cooldown.

The paper presents the RELAP5/mod3.3 analysis of natural circulation cooldown with one inactive loop for Nuclear Power Plant Krško (NEK). The aim of the analysis is to determine the limiting cooldown rates during operator recovery actions to minimize the effect of flow stagnation in inactive loop.

Since this is typical asymmetrical transient, the RELAP5/mod3.3 NEK model with split reactor vessel model was developed (models of the reactor vessel and core were axially divided in two parts) and used for this analysis.

The several transients of cooldown, with one inactive loop, for different time after shutdown (different decay heat) were performed. The extreme conservative assumptions were applied for the analyses, i.e. the complete loss of feedwater (FW) and auxiliary feedwater (AF), including turbine driven AF pump, and the cooldown has started after the SG is completely dry (inactive).

The analyses show that the cooldown rate shall be significantly reduced, and, based on the results the procedure ES-0.2 "Natural Circulation Cooldown" was modified.

Keywords: *natural circulation, inactive loop, cooldown rate, RELAP5/mod3.3, Nuclear Power Plant Krško (NEK)*

S1-134

Thermohydraulic Instabilities in a Parallel-Tube System

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Thermohydraulic instabilities are very common phenomenon in fluid flow systems. They may occur in any flow regime, in any system, under natural or forced circulation, but their special group are two-phase instabilities. This type of instabilities occur in a two-phase flow regime or in a single-phase flow regime close to the saturation state. In every case, oscillations of flow rate and system pressure are undesirable [1]. Their possible consequences are: reactor control problems, an elevated risk of heat transfer surface burnout, thermal waves and mechanical vibrations.

In this paper, flow oscillations at the phase transition are addressed. A loop system consisting of parallel tubes is simulated in the SIMMER-V code. SIMMER-V is a thermohydraulic code originating from AFDM multi-phase CFD software. SIMMER-V is currently developed at the French Alternative Energies and Atomic Energy Commission (CEA) and the Japan Atomic Energy Agency (JAEA) in the frame of the joint France-Japan R&D collaboration. It is mainly used for SFR calculations with the main focus on severe accident loss-of-flow sequences.

For multiphase calculations, SIMMER-V uses the popular technique of regime mapping to represent the flow topologies on a void fraction/entrainment map [2]. There are few validation studies that include the specific situation of two-phase instability and behavior at the two-phase flow regime boundaries. According to the driving mechanisms of different oscillation types, one can differentiate static and dynamic oscillations, when the later constitute the major group of oscillations is real systems. In such systems, different types of oscillations are favored at certain range of the boundary conditions: heat flux, mass flow rate, inlet subcooling and local flow restrictions.

[1] S. Kakac and B. Bon, "A Review of two-phase flow dynamic instabilities in tube boiling systems," Int. J. Heat Mass Transf., vol. 51, no. 3, pp. 399–433, Feb. 2008

[2] S. Kondo, H. Yamano, and T. Suzuki, "SIMMER-III: A computer program for LMFR core disruptive accident analysis. Version 2. H model summary and program description," Japan Nuclear Cycle Development Inst., JNC-TN--9400-2001-002, 2000

Keywords: *fluid flow, instabilities, two-phase flow, SIMMER-V*

S1-135

RELAP5 Simulation of Design Extension Condition with Loss of All Feedwater in PWR

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After Fukushima Dai-ichi, the design extension conditions (DECs) were introduced by Western European Nuclear Regulators Association (WENRA) reference levels, following International Atomic Energy Agency (IAEA) definition. Slovenia implemented WENRA reference levels, which required to consider design extension conditions (DEC). For both, WENRA and IAEA, the total loss of feedwater event can be considered as a Design Extension Condition (DEC) event without significant fuel degradation (WENRA classifies it as a category DEC A).

The purpose of this study is to determine whether the total loss of feedwater had to be considered as DEC A in a specific two-loop pressurized water reactor (PWR). Namely, the control of DECs is expected to be achieved primarily by features implemented in the design (safety features for DECs) and not only by accident management measures that are using equipment designed for other purposes. The selected total loss of feedwater DEC has been derived from probabilistic safety assessment (PSA) according to IAEA recommendations. The initiating event is the loss of all feedwater. This means that besides the loss of main feedwater also both motor driven auxiliary feedwater (AFW) pumps and the turbine driven AFW pump are assumed to be unavailable. An additional operator action is reactor coolant pump trip according to the emergency operating procedures (EOP). The plant is initially at 100 % reactor power (2000 MWt). Both trains of high pressure safety injection (HPSI), low pressure safety injection (LPSI) and accumulators are assumed to be available. The RELAP5/MOD3.3 Patch 5 computer code and the standard input deck for selected two-lop PWR has been used for calculations.

The simulation results of total loss of feedwater showed that a new DEC safety feature would be needed. Namely, the existing HPSI pumps could not inject into the primary system due to the high primary pressure during total loss of feedwater. The primary pressure depends on the primary side relief valves setpoints. Without considering the new DEC safety feature (or accident management measures to depressurize the primary system), the total loss of feedwater leads to overheating and damage of the core in about one hour.

Keywords: *total loss of feedwater, RELAP5/MOD3.3, design extension condition, pressurized water reactor, safety analysis*

S1-139

Thermal-Hydraulics Analysis of the IAEA CRP FFTF LOFWOS Test#13

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Global interest in fast reactors has been growing since their inception in 1960 because they can provide efficient, safe and sustainable energy. Their closed fuel cycle can support long-term nuclear power development as part of the world's future energy mix and decrease the burden of nuclear waste.

Within this framework, the IAEA organized a Coordinated Research Projects (CRP) on FFTF Loss of Flow Without Scram (LOFWOS) Test #13, aimed at improving Member States' fast reactor analytical simulation capabilities, international validation, and qualification of codes currently employed in the field of fast reactor.

The Fast Flux Test Facility (FFTF) was a 400 MW thermal powered, oxide-fueled, liquid sodium cooled test reactor built to assist development and testing of advanced fuels and materials for fast breeder reactors.

The present paper shows the work performed by NINE for the CRP focused on benchmark analysis of one of the unprotected passive safety demonstration tests performed at the FFTF. In particular, a detailed nodalization was developed following the NEMM (NINE Evaluation Model Methodology) already applied for LWR safety analysis. After achievement of acceptable steady-state results, transient analysis was performed. In addition, the NINE validation procedure was adopted in order to validate the Simulation Model (SM) against the experimental data. Two system thermal-hydraulic codes, namely RELAP5 and TRACE, were used to analyse the selected test and the comparison between the two SM results is also presented in this paper.

The final goal of the activity is to present the main outcomes achieved through the use of codes currently employed in the field of fast reactor, and how the application of the NEMM procedures allows to develop and qualify the SM results and validate the computer codes against experimental data.

Keywords: *FFTF, LOFWOS, RELAP5, TRACE, NEMM*

S1-140

Simulation Of The OECD/NEA Rod Bundle Heat Transfer (RBHT) Benchmark With RELAP

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The OECD/NEA RBHT (Rod Bundle Heat Transfer) Project is an International three-year NEA Joint Project whose objective is to conduct new experiments and evaluate system hydraulics and sub-channel codes in the simulation of reflood tests. Such tests are performed in a full height rod bundle facility equipped with advanced instrumentations capable to measure the real-time droplet field, cladding and steam/fluid temperatures, water carryover fraction and pressure drops. The test matrix encompasses both steady and oscillatory reflood inlet flow conditions. Within the RBHT project, a challenging benchmark exercise is conducted, including an open and a blind test phase providing a unique opportunity to project's participants to validate codes and nodalization techniques. This paper presents a validation study of the RELAP code on the RBHT open test series. The simulations' results generally well agree with the measured data, according to the accuracy metrics proposed by the benchmark team. A larger discrepancy is detected for experimental tests characterized by higher flooding rates with low subcooling degree. Several model's parameters have been investigated including also different nodalization schemes to characterize the impact on the predicted results during the sensitivity analysis.

Keywords: *Validation, Reflood, Accuracy, RELAP*

S1-147

NEK 3 inch Cold Leg Break LOCA Calculation using TRACE 5.0p5 and RELAP5/MOD 3.3 Codes

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NPP Krško input deck for best-estimate computer code TRACE is being developed at Faculty of Electrical Engineering and Computing (FER), Zagreb. The TRAC/RELAP Advanced Computational Engine (TRACE) is an advanced, best-estimate reactor systems code developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady state behavior in light water reactor.

At FER, a detailed RELAP5/MOD 3.3 model for NPP Krško has been developed. The model encompasses detailed models of control and protection systems, e.g., Automatic Rod Control system, Safety Injection System, pressurizer pressure and level control system, steam generator level control system, steam dump control, etc. The model is being constantly upgraded in accordance to plant modifications.

The RELAP5/MOD 3.3 input deck for NPP Krško was used as a basis for development of NPP Krško model for TRACE 5.0p5 code. In the paper the results of steady state qualification (first 1000 s) for both RELAP and TRACE are assessed against plant referent data. For on-transient qualification of a developed TRACE model, 3 inch small break LOCA accident in a loop with pressurizer was analyzed and the results were compared with RELAP5 results. In the analysis all the Engineering Safety Features (ESFs) with a minimum delay were assumed available. The transient was simulated for 10000 seconds.

Keywords: *TRACE, RELAP5, steady state qualification, on-transient qualification, small break Loss of Coolant Accident*

S1-150

Influence of reactivity feedback modelling in RELAP5 NEK LONF ATWS calculation

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Standard NPP Krško input deck for RELAP5/mod 3.3 code was used in calculation of LONF ATWS accident with and without AMSAC actuation. The influence of fuel temperature reactivity, moderator temperature reactivity and boron concentration to peak primary pressure and core thermal hydraulic conditions was analyzed for BOL, MOL and EOL core. The main intention is to evaluate acceptability of current simplified boron feedback model in case of low coolant density in the core or partial core uncover.

Keywords: *RELAP5, NEK LONF ATWS, reactivity feedback, boron reactivity, core uncover*

S1-154

Impact of Selected Long-Term Operation Improvements Relevant to the Pressurized Thermal Shock in PWR

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This paper presents the impact of six different long-term operation (LTO) improvements relevant to the pressurized thermal shock (PTS) phenomena in a pressurized water reactor (PWR). Four LTO improvements are connected to the change in the specific parameters of the reactor safety systems: a heating of water in high-pressure injection (HPI) tanks, a heating of water in accumulators, a heating of water in low-pressure injection (LPI) tanks and a decreasing the accumulator pressure. The remaining two LTO improvements are related to human factors and they are modeled as different operator actions: a reduction of high-pressure injection system (HPIS) flow by the operator and a different secondary side cooldown rate.

The thermohydraulic (TH) impact of the LTO improvements is studied with the RELAP5/Mod3.3 computer code with attention on the reactor pressure vessel that is vulnerable to the PTS. The 2-dimensional nodalization is applied to reactor downcomer section to enable modelling of asymetric cooldown of the reactor pressure vessel (RPV). The sequence selected as relevant for PTS analysis is the small break loss of coolant accident (SB-LOCA) with a 50 cm^2 break in the hot leg (HL) and coincident with a loss of offsite power. The studied plant used in the analysis is based on a 1300 MW four-loop PWR German design. Analyses show that the most promising results for the SB-LOCA are obtained for the heating of water in HPI tanks and the reduction of HPIS flow by the operator.

The presented work is a part of the larger benchmark performed in the Work Package 2 of the APAL project (Advanced PTS Analysis for LTO) project in the frame of the Euratom research and training programme.

This work is focused on the TH effects of the LTO improvements. Further deterministic and probabilistic fracture mechanics benchmarks will be performed in APAL to quantify the effect of selected LTO improvements on final fracture mechanics results.

Keywords: *pressurized thermal shock, PTS, long-term operation improvement, thermohydraulic analysis, RELAP5.*

S1-182

Performance Based Risk Informed Fire Modelling Evaluation of Electrical Equipment Functionality in Nuclear Power Plants

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Introduction of risk-informed and performance-based analyses into fire protection engineering practice exists in both the general fire protection and the nuclear power plant fire protection applications. Risk-informed and performance-based approach relies on application of validated and verified fire modelling to estimate fire generated effects that are arising in predefined fire scenarios for fire protection related applications in nuclear power plant. Regulatory bodies have used risk-informed insights as a part of its regulatory decision making for the past thirty years. Before performance-based approach came out, all regulatory prescribed requirements relied on deterministic approach with ultimate condition that one complete shutdown train together with auxiliary support features is free of fire damage. Performance-based approach relies upon calculable performance results to be met but provides more flexibility in achieving established performance criteria during all phases of plant operations. Nevertheless, fire modelling is finding its benefits in design basis engineering, fire hazard analysis, nuclear safety capability assessment and probabilistic risk assessment. To demonstrate such capabilities, an example on fire development in Nuclear Power Plant Safety Related Pump Room with respect to possible loss of one safety shutdown path is modelled with a fire simulator computer tool.

Keywords: *fire, fire model, verification, validation, nuclear, fire protection, deterministic, risk-informed, performance-based, experiment, electrical equipment*

Session 2
Operation and Maintenance Experience (OME)

S2-102

Review of the Operating Experience with the Surveillance Testing of Passive Autocatalytic Recombiners in NPP Krško

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Passive Autocatalytic Recombiner (PAR) system was installed in NPP Krško to mitigate design basis (DBA) and design extension condition (DEC) accidents by minimizing the risk of containment damage due to hydrogen deflagrations. The mitigation strategy is based on the catalytic oxidation of hydrogen (H_2) using oxygen (O_2) from the containment atmosphere and a noble metal palladium (Pd) as catalyst.

Every 18 months the periodic testing of PAR cartridges is performed. The purpose of the PAR cartridge surveillance test is to verify the system operability and assess the need for cartridge regeneration. The testing of the PAR cartridges has shown fluctuations in the results, which could have been caused by testing method sensitivity to environmental conditions, impurities in the containment atmosphere and/or material properties (e.g., amount of catalyst, surface area of reactants, heat addition, etc.).

The issues with the periodic testing had no impact on the system performance. It has been shown that the selection of the testing parameters and the test device design have not been correctly defined. This is confirmed by the results of various tests performed in the period 2015 – 2021. Additionally, operating experience from different PAR users and suppliers indicated the conduction of test at elevated temperatures. NPP Krško insisted, with the supplier of the testing device to solve these issues. The joint approach was to increase the test temperatures to those, which are closer, yet still conservative, to accident conditions (approx. 40 °C). Under accident conditions the temperature of the containment atmosphere would be much higher than the normal ambient containment temperature. Testing performed at elevated temperatures reduces the impact of the known environmental influences on reaction start-up time.

Numerous tests and design modifications of the test device itself have been made, with the aim of performing the tests at elevated temperature.

The paper reviews NPP Krško's operating experience with the testing of PAR system and the activities performed in order to eliminate difficulties with the surveillance testing.

Keywords: *passive autocatalytic recombiners, PAR, periodic testing, DBA, DEC, PAR testing device, hydrogen*

S2-106

Assuring Shutdown Safety at Krško NPP

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Nuclear power plant operating experience has shown that PWRs are susceptible to a variety of events during shutdown conditions. While these events have caused no harm to the public, nor posed an undue risk, they do illustrate that careful attention to outage management and shutdown operation is necessary to maintain adequate defence-in-depth for safety functions.

Many activities including generic communications, site visits, workshops, studies and surveys conducted by the Institute of Nuclear Power Operations (INPO), the Electric Power Research Institute (EPRI), and the World Association of Nuclear Operators (WANO), have heightened the awareness of shutdown concerns. This awareness is a prerequisite to enhancing shutdown safety.

The scope of activities that each utility undertakes during a normal refuelling outage is large and diverse. Besides refuelling, activities associated with preventive and corrective maintenance, modifications, surveillance testing, in-service inspection, and the administrative activities that support these tasks make outage planning and control a significant challenge. The coordination of these activities with the objective to manage risk and maintain key safety functions is essential and goes beyond compliance with technical specifications requirements during shutdown. In addition, while the scope of activities for an unplanned or forced outage is far less than that of a refuelling outage, the same awareness of vulnerabilities during shutdown conditions is required to safely conduct these outages.

The critical outage safety functions at Krško NPP are reactivity control, decay heat removal, reactor coolant inventory control, electrical power availability, spent fuel pit cooling, support systems availability and containment integrity. Those functions provide baseline for creating the outage plan. Activities planned for the upcoming outage contain Outage Risk Assessment and Management (ORAM) codes which serve as input data for Krško NPP Outage deterministic and PSA evaluation using software tool PARAGON. Results of the calculations are Core Damage Frequency, Core Damage Probability and the status of the equipment required to fulfil critical safety functions during shutdown.

As an example, preparation and execution of Krško NPP 2019 and 2021 outages are shown as well as how is the monitoring of the critical safety functions being performed during outage execution.

Keywords: *shutdown safety, outage, planning, equipment protection*

S2-116

Mechanical and 1st Chemical Cleaning of NEK Steam Generators

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Sludge removal is performed on two steam generators (SG's) at the Krško Nuclear Power Plant (NEK) during every outage. SG's are a meeting point of five major plant systems: Reactor Coolant System (RC) on the primary side and four systems on the secondary side – Auxiliary Feedwater System (AF), Blowdown System (BD,) Main Feedwater System (FW), and Main Steam System (MS).

Sludge removal activities take place on the secondary side of the SG's on the top of the tube sheet. It always consists of classical Sludge Lancing (SL) which is done by spraying water at different angles (30°, 90°, 150°) between the tube gaps in the steam generator tube bundle with a pressure of around 220 bars. Another method is Inner Bundle Lancing (IBL) which means spraying water directly inside the tube bundle by a traveling lance tape with a spraying nozzle at the end. Water is sprayed at an angle or directly on the top of the tube sheet with a robot-guided manipulator which is placed inside a steam generator. The manipulator and therefore the spraying action is controlled by an operator and at times it is fully autonomous to provide the highest protection measures possible. Another method of sludge removal which was for the first time utilized in 2019 at the Krško site was Chemical Cleaning (CC) of both SG's. During this process, a chemical was injected into the SG's through the BD system and periodically pumped between the two to create a dynamic flow and maximize the cleaning effect. In order to achieve the best results, a constant temperature of the chemicals had to be maintained at all times. Upon completion of chemical cleaning, a rinsing phase was followed to remove any post-treatment chemicals. After all sludge removal activities, a televisual inspection (TVI) of the top of the tube sheet was performed to access the hard sludge area and to search for potential foreign objects in the SG's. If for instance an object of importance during TV inspection is found, an attempt to retrieve it would usually take place. Other methods of sludge removal such as upper bundle flushing or ultrasonic cleaning have not been implemented in NEK thus far.

Since the power plant uprate in May 2000, NEK conducted SL on both SG's every outage also starting with IBL in 2013 and 2015, and the same method was used in the 2018 outage. During the last outage in 2019, all three methods (SL, IBL, and CC) have been utilized with the main purpose to extend the full load operation of the plant, prevent and/or stop denting processes in the SG's from occurring, reduce and stop the build-up of hard sludge area to increase/sustain efficiency and remove foreign objects found in the SG's.

SG's U-tubes are a barrier between the primary side coolant and the secondary side of NEK and the environment. Therefore, it is crucial to keep the highest level of integrity of the U-tubes because any leak could potentially mean a release of radioactive material to the atmosphere.

This paper describes the purpose and workflow of sludge removal activities in the outage of 2019 in NEK.

Keywords: *Steam Generators (SG's), Sludge Lancing (SL), Inner Bundle Lancing (IBL), Chemical Cleaning (CC), televisual inspection (TVI), Foreign Object Search and Retrieval (FOSAR), Krško Nuclear Power Plant (NEK)*

S2-119

Corrosion Detection and Surface Repair with Coatings on Condensate Storage Tanks Internal Surfaces

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In the Nuclear power plant Krško, there are two single hull condensate storage tanks with floating diaphragm each containing up to 757 m^3 of demineralized water. The main purpose of these two storage tanks is to provide a capacity of cooling water for cooling the reactor coolant system via steam generators with the use of auxiliary feedwater pumps. This is a very important function from the safety point of view and that is the reason that both storage tanks are listed as safety class 3 components. It is also possible to fill up the condensate system if other means are not available. Condensate storage tanks are subject to periodic testing and periodic inspections to determine the state of the components. Both tanks are in operation for more than 35 years, are located outside, and are exposed to different degradation processes. There was a concern that the tanks are leaking because there were often small puddles of water near the tanks. There were no changes in the levels of tanks. The design of tanks is a single hull, so there is no indication if a small leak is present.

In the outage 2019, both tanks were emptied and examined with NDE methods to find any corrosion damage of floor plates and any untight spots on adjacent welds.

The article is about the NDE methods that were used (Magnetic Flux Leakage, Ultrasonic and Vacuum Box inspection) to determine the condition of the floor plates and adjacent welds as well as the process of internal surface reparation with coatings. Process of coatings qualification for use in safety class components is also explained: dedication process for material up-grade from non-safety related to safety-related because CY tank linings are classified as safety-related according to RG 1.54, rev 2 and corresponding ASTM standards and NEK technical specification SP-A5001. All activities for surface repair with coatings shall comply with safety-related requirements. Also, extensive immersion tests with selected and specially defined parameters were performed in NEK chemical laboratory in order to select the most suitable coating system for surface repair of CY tank floor lining. Further details concerning immersion tests are presented below.

Keywords: *Corrosion, Condensate Tank, NDE, Coating, Aging Management*

S2-123

TARGET – Development of Submersible ROV System for BMN Inspection

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Most PWRs have penetrations in the RPV lower heads for in-core nuclear instrumentation. These penetrations generally are made of nickel-based Inconel Alloy 600. Weld materials are typically Alloy 82/182. Operating conditions of PWR plants are causing nickel-based alloys cracking through a process called primary water stress corrosion cracking (PWSCC). In 2003, the licensee for the South Texas Project Unit 1 (STP-1) identified apparent boron deposits on the lower RPV head near two bottom mounted nozzles (BMNs). The NRC issued Bulletin 2003-02 to obtain information on licensee inspection activities and inspection plans for the RPV lower head. EPRI issued MRP-206 report that provides inspection and evaluation guidelines for BMNs for PWR plants, including guidelines for periodic bare metal visual examination for evidence of primary coolant leakage, or periodic non-visual nondestructive examinations for indications of service-induced cracking. The non-visual inspections (ultrasonic testing examination) may detect service-induced degradation before through-wall cracking, leakage, circumferential cracking below the bottom of the J-groove weld, release of loose parts, or incipient boric acid wastage of the low-alloy steel reactor vessel lower head material occurs. Therefore, periodic examinations will adequately manage potential for cracking by PWSCC and preserve structural integrity. INETEC developed TARGET system for BMNs inspection, consisted of submersible ROV and specially designed probe, composed of several UT probes. UT system and technique to detect, length and depth size the service-induced degradation in the BMN volume material is developed. The EPRI NDE Center performed a technical review and validated INETEC's ultrasonic examination technique for BMNs. Aforementioned validation was done according to requirements defined by: 1) MRP-206, 2) MRP-411.

Keywords: *Submersible ROV, BMN inspection, nondestructive examination, ultrasonic testing examination, ultrasonic examination technique*

S2-124

Development of conforming ultrasonic probe for inspection of ITER experimental reactor

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This paper presents the development of an ultrasonic probe for inspection of welded joints between two, 60 mm thick, sector plates of the ITER (International Thermonuclear Experimental Reactor) vacuum vessel. Several points were raised during the analysis of the welded joint specification and environmental conditions that limit the use of conventional UT probes for this purpose. These points include inspection of the weld from the weld root side that's unmachined, significant weld thickness of 60 mm and medium around weld being air. The development consisted of three mutually dependent parts, of which the first one is the choice of flexible material adequate for conforming to complex surfaces. Parallel to this, it was necessary to simulate the ultrasonic signal to determine probe geometry considering the weld dimensions and specifications of the material AISI 316L(N)-IG being inspected. Finally, the couplant problem was approached. Without couplant, air trapped between conforming material and inspected material prevents the propagation of the ultrasound waves. Due to ITER vacuum restrictions, it was imperative to develop a solution that uses the minimum amount of couplant that still provides a satisfactory ultrasonic signal. The development process included computational analysis, probe prototyping, test blocks manufacturing and experimental tests. Once developed, the probe was subjected to comprehensive tests on qualification blocks matching inspection objects with intentionally implemented artificial defects.

Keywords: *ITER project, Non-destructive testing (NDT), Ultrasonic Testing (UT), UT probe development*

S2-128

Improvement possibilities for nuclear power plants inspections by adding deep learning based assistance algorithms into a classic ultrasound NDE acquisition and analysis software

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The safety of nuclear power plants has always been one of the most important security issues in industry in general. Numerous standards, techniques and tools have been developed to deal specifically with the safety of nuclear power plants – one has specialized probes, robotized systems, electronics and software. Although seen as a mature (or slowly evolving) industry, this notion about the nuclear safety is a bit misleading – the area is developing in many promising new directions. Some recent global events will speed up this development even more.

On the other hand, industry is currently going through digital transformation, which brings networking of devices, equipment, computers, and humans. This fourth industrial revolution promises speed, reliability, and efficiencies not possible up until now. In the NDE sector, new production techniques and traditional manufacturing lines are getting to be lights-out operations (near total automation). The same is most probably going to happen with the safety inspections and quality insurance. Robotics and automation are improving worker safety and reducing human error. The well-being of inspectors working in hazardous environment is being taken care of. Most experts agree that the digitalization of NDE offers unprecedented opportunities to the world of inspection for infrastructure safety, inspector well-being, and even product design improvements. While the community tends to agree on the value proposition of digital transformation of NDE, it also recognizes the challenges associated with such a major shift in a well-established and regulated sector.

The work presented in this paper shows a part of the project that aims to develop a modular ultrasound diagnostic NDE system (consisted of exchangeable transducers, electronics and acquisition/analysis software algorithms), for applications in hazardous environments within nuclear power plants. The paper will show how the software part of this system can reach near total automation by implementing various deep learning algorithms as its features and, then, testing those algorithms on laboratory samples, showing encouraging results and promises of online monitoring applications.

Furthermore, future general prospects of this technology are discussed and how this technology can affect the well-being of nuclear power plants inspectors and contribute to overall plants safety.

Keywords: *ultrasound, nuclear, safety, deep-learning, industry 4.0*

S2-162

Evaluation of Over Temperature Delta T and Over Power Delta T Operating Margin in Krsko NPP

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The purpose of the reactor protection system is to prevent reactor conditions from exceeding the safety limits by initiating reactor trip. Delta T protection system is designed with assumption that core power is, to a good approximation, proportional to the temperature difference between vessel outlet and inlet. Since the core power level is important parameter in determining when the core is approaching departure from nucleate boiling (DNB) or fuel meltdown, Delta T protection system is used to protect the core against those accidents.

Over Temperature Delta T protection (OTDT) is designed to protect the reactor core against departure from nucleate boiling (DNB) and prevent overheating of the fuel rod cladding. Measured DT signal is compared with continuously calculated OTDT setpoint and if actual Delta T exceeds OTDT setpoint the OTDT protection will generate a reactor trip signal. OTDT setpoint is a function of average RCS temperature, pressurizer pressure and axial flux difference.

Over Power Delta T protection (OPDT) is designed to protect the fuel from reaching melting point and prevent fuel rod cladding failure. Measured Delta T signal is compared with continuously calculated OPDT setpoint and if actual Delta T exceeds OPDT setpoint the OPDT protection will generate a reactor trip signal. OPDT setpoint is a function of average RCS temperature only.

This paper will show how margin between measured DT and calculated OTDT and OPDT setpoints change with the change of different plant parameters (average RCS temperature, pressurizer pressure and axial flux difference).

Keywords: *OTDT protection, OPDT protection, DNB, fuel meltdown, operating margin.*

Session 3
Nuclear Energy: Planning, Environment and Technologies (NEPET)

S3-104

Global Warming Effect of Switchgear Using Sulphur Hexafluoride in Nuclear Power Plants

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As we are all aware that nuclear power plants are in vast majority made as part of the electric power grid system. Therefore, it is unavoidable that a switchyard plays a vital role as part of power plant operation and nuclear safety. Historically, to reduce size and increase reliability, gas SF_6 – sulphur hexafluoride has been noted and widely used as the most successful medium for reliably breaking the electric current. Sulphur hexafluoride can be found in basically all high-voltage equipment such as circuit breakers, voltage and current transformers and gas insulated substations. SF_6 gas is the most potent greenhouse gas with 3,500 years of decay half-life in nature cycle and 22,800 times more impact on global warming than CO_2 has. Typical switchyard element has around 5–15 kg and GIS bay up to 170 kg of this gas. That would mean 456 tons and 3,876 tons of equivalent CO_2 respectively, if a gas would completely escape to atmosphere. By the content of SF_6 all mentioned elements so far are easily overshadowed by the generator load break switch whose task is to add an additional layer of protection to the generator. A comparison of results with CO_2 equivalent will be presented and converted to information, easily understood by the general public, for instance the number of kilometres needed to be made by car to create the same amount of emissions and reach the same global warming potential – GWP. The EU f-gas regulation and its future will be presented, as well as how it influences using such gas in switchgear elements, since long-term EU plans are to ban or heavily tax the use of technologies presented in f-gas regulation, where a sufficient alternative technology level is reached. The article will also cover alternatives that are available in the market, their strengths and weaknesses, including their influence on nuclear safety and costs related to using alternative gas mixtures.

Keywords: *Global warming, GWP, SF_6, CO_2, Circuit Breaker, F-gas, EU Regulation, Switchyard.*

S3-109

Economic Modelling of New Nuclear Power Plants

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In the 21st century the world faces a new challenge of drastically reducing emissions of greenhouse gases while simultaneously expanding energy access and economic opportunity to billions of people. Nuclear energy would seem to be ideally placed to meet that challenge.

Nuclear energy's future role, however, is highly uncertain for several reasons: primarily, escalating costs and, to a lesser extent, the persistence of historical challenges such as spent fuel disposal and concerns about nuclear plant safety and nuclear weapons proliferation.

Nuclear power plants are expensive to build but relatively cheap to run. In many places, nuclear energy is competitive with fossil fuels as a means of electricity generation. Waste disposal and decommissioning costs are usually fully included in the operating costs. If the social, health and environmental costs of fossil fuels are also taken into account, the competitiveness of nuclear power is improved.

The basic metric for any generating plant is the levelised cost of electricity (LCOE). It is the total cost to build and operate a power plant over its lifetime divided by the total electricity output dispatched from the plant over that period, hence typically cost per megawatt hour.

This paper will describe the development and results of two economic models of new nuclear power plants using the most recent and relevant data available to the authors.

Keywords: *NPP costs estimates, LCOE, economic calculations, economic modelling*

S3-114

Utilization of Waste Heat From Gas Cooled SMR in Water Desalination

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The Gas-Turbine Modular Helium Reactor (GT-MHR) incorporates two-stage compression with precooling and intercooling in order to compress the helium to higher output pressures thus achieving high-pressure ratios across the compressors. Doing so, minimizes the total work consumed by the compressors. Both the pre-cooler and intercooler facilitate this process by reducing the inlet temperatures of the working fluid; in fact, they are specialized heat exchangers that use water to cool the helium before entering each of the compressors. At the optimum pressure ratio of PR_C = 3.025, and the optimum turbine inlet temperature of T_1= 850 ^{0}C; the coolers will then dissipate heat at a rate of about 318 MW. An energy analysis is conducted to utilize this waste heat for the desalination of seawater by a process called Seawater Freezing Desalination (SFD). This is achieved by connecting an Organic Rankine Cycle (ORC) to produce useful power. The power produced by the ORC is supplied as electrical power to the compressor of a vapor-compression Refrigeration Cycle (RC) via an electric generator. The connection between the RC and the SFD cycle is established through the evaporator of the RC. The optimum parametric values of the new cycle are obtained by using the Engineering Equation Solver (EES) software. It was found that the efficiency of the combined GT-MHR/ORC cycle was higher than the simple GT-MHR by 2.5%; reducing the waste heat by 61.4%, and producing around 1.507 Million Cubic Meters (MCM) of freshwater annually.

Keywords: *Gas cooled Modular Reactor, Nuclear Freezing Desalination, Organic Rankine Cycle, Cogeneration, Refrigeration Cycle*

S3-118

A Comparison of the Radioactive Waste Produced for Different Nuclear Energy Development Scenarios

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Climate change is increasingly affecting humanity. Electrical energy generation based on nuclear technologies can compete with the other energy generating technologies due to constant capacity factors and low greenhouse gas (GHG) emissions. The specific GHG emissions of nuclear power plants are among the lowest of any electricity generation method. Global primary energy needs rise more slowly than in the past, but still an increase of 25% between today and 2040 is expected according to World Energy Outlook 2018. Electrical energy needs will rise faster than primary energy needs and the electrical energy production has to be with low GHG emissions due to global warming mitigation. We assume in our scenarios that nuclear energy will be global electricity production leader with a percentage of 34,1 % in the year 2040. In addition, we assume that all thermal power plants will be replaced by uranium or thorium fuel cycle nuclear power plants by the year 2056. This paper describes a comparison of three different long term nuclear energy development scenarios according to production of radioactive waste. In the first scenario only pressurized water reactors (PWR) power plants will be used by the end of this century. In the second scenario PWR power plants will be used until 2050 and after that year fast breeder reactor (FBR) nuclear power plant will be introduced gradually. In the third scenario PWR power plants will be used until 2050 and after that year molten salt thorium reactors (MSTR) power plants will be introduced gradually. These scenarios are compared according to production of radioactive waste (volume and activity).

Keywords: *Climate change, GHG emissions, Fast breeder reactors, Thorium fuel cycle, Radioactive waste*

S3-120

Increasing the Efficiency of NPP by Using the Heat Pump for Heat Supply

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NPPs and TPPs even in condensing mode are used for heat supply during winter. The power plant heats up network water using a steam from turbine. The possibility of using a waste heat of NPP for heat supply with a heat pump (HP) are considered.

Analysis the efficiency of the HP to use a low potential heat from cooling water of steam condenser was performed. Developed an algorithm for calculating the thermal pump was developed taking into account the properties of the refrigerant, which is ecologically acceptable. Two options for the location of the evaporator HP are considered: directly in the condenser and in the form of a separate heat exchanger. The dependence of the coefficient of performance (COP) from the pressure in steam condenser is evaluated.

The economic effect of the HP using due to the evaluation of possible profits from the production of an additional amount of electricity is analysed. When using HP, the heating plant does not consume a steam for heating network water. Electricity production, taking into account the power of the compressor, may increase. The additional amount of heat supply makes the use of HPs economically justified.

Keywords: *heat supply, heat pump, coefficient of performance, NPP efficiency*

S3-126

SMR application into existing infrastructure and new localities in the Czech Republic

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Emerging development of SMR project led to widespread interest in application of those technologies in the Czech Republic in past years. In early 2010's SMR projects were mostly of marginal interest amongst country power plants operators or government projects. UJV started first cooperations with SMR projects in 2013 (NuScale, MPower) as a part of government project for utilization of SMR in the Czech Republic. Initial results combined with promising investment estimations were suppressed by immaturity of design and uncertainties of first of a kind licensing.

Growing interest in alternative energy solution to coal and other fossil sources which will be terminated in next 20 years and major progress of SMR projects brought back interest in SMR.

As CEZ (Czech energy operator) push more efforts in SMR in starting in 2019 with initial pre-feasiblity study, UJV countinues with SMR qualification study in 2020/2021 and follow up in 2022- 2024 period with Feasibility study for selected "nuclear locality" where NPP is already in operation and study for 6 existing coal localities with goal to selection 1-2 localities for more detailed pre-feasibility study. These studies are closely related to locality characteristics and its suitability for nuclear utility as SMR.

General public knowledge about SMR brought interest also to municipalities which are currently based on fossil energy and heat sources. Based on this interest UJV performed major study for Moravia – Silesian part of the Czech Republic. Besides the above-mentioned activities UJV is also part of projects organized by Technology agency with topics focused on general applicability of SMR in in energy sector of the Czech Republic.

The presentation will describe the following major topics performed in UJV SMR studies:

- Technical Review of Vendor's data
- Licensing
- Road maps of implementation
- Site surveys based on regulatory requirements and limitations
- Economic and financial analysis
- Methodology for multicriteria evaluation of a small nuclear modular reactor (SMR) concept

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

S3-127

A Strategy for Deployment of Thorium and U233 (In Italy)

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Thorium is an unavoidable nuclear fuel generator and a gigantic energy source, should nuclear technology become a strategy for the electricity production in regions like EU or in Countries like Italy. The history of exploitation of thorium in nuclear technology provides a number of dead-end endeavors; at the same time, a large number of initiatives and ideas are fluorescent in various parts of the world dealing with the deployment of thorium as nuclear fuel.

After examining the history and the current trends, the present paper deals with a dream project (for Italy): the U-233 extracted from a thorium breeder reactor is stored to constitute an energy deposit, suitable for decades energy consumption, which has the potential to make stable the energy market, without targeting electricity production in the Country. Three topics touched in the paper are: a) U-233 generation details; b) chemical separation of fissile and fertile materials; c) challenges for a nuclear reactor to produce electricity, desalinated water and fissile material, simultaneously.

Keywords: *Strategic Energy Reservoir, Thorium Fuel, Uranium-233, CANDU Reactor*

S3-138

Why Nuclear Power is Essential for Reducing Emissions of Greenhouse Gases

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The environment, climate change and contribution of the greenhouse gas emissions to global warming became a worldwide issue. It is recognized that energy and transport sectors are major contributors to the emissions and therefore focus should be given mainly in these sectors to the actions reducing the emissions and thus contributing to carbon neutrality.

It is obvious that nuclear power has the potential to address the issue by sustainably and reliably supplying the large quantities of clean and economical energy needed to run industrial societies with minimal emission of greenhouse gases. Unfortunately, some EU member states strictly object the nuclear power and insist on the use of solely so-called renewable power sources, in particular solar and wind power. Societal and financial support to such sources continues and is even further strengthened at present in spite of their questionable benefits and contradicting the exclusive rights of each country to decide on proper energy mix.

The paper will first summarize up-to-date facts demonstrating the severity of the issue worldwide and in EU in particular. Further on, the problems associated with the use of intermittent power sources such as solar and wind will be discussed. Sustainability and potential contribution of intermittent sources to reduction of the emissions will be questioned. Current status of the energy storage options and other possibilities to address the intermittency will be briefly summarized.

Other environmental impacts caused by opposing the use of nuclear power and non-critical support to renewable sources will be also presented. Comparison of various power plants from different viewpoints will be made such as consumption of construction materials, land requirements, production and disposal of waste, induced mortality rate and safety.

Keywords: *GHG emissions, environment, sustainability, renewables, nuclear power plants*

S3-142

Development of Hybrid Gamma-ray Spectrometry Methods for Enhancing the Capacity of Environmental Radiological Monitoring Around Nuclear Power Plants

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In-situ gamma spectrometry allows complete characterization of gamma radiation fields at a given location through accurate and sensitive determination of the concentration of radionuclide activity in the soil. When compared with methods that include sample collection and subsequent laboratory analysis, the in-situ method enables better soil representativeness, direct and accurate determination of areas of interest (hot spots), higher speed of measurements and reduced monitoring costs. All this makes it a very important complementary or alternative method to laboratory measurements, and unavoidable in case of accidents. The primary goal of this work is to present concepts behind a development and implementation of hybrid gamma spectrometry method for radiological monitoring of the environment of nuclear power plants. The hybrid method would integrate an innovative approach in determining areas of interest with the newly developed method for in-situ gamma spectrometric measurements. This would increase the speed of radiological survey, which is critical for rapid decision-making, and in turn would enable the implementation of surveillance of a larger area. For the purpose of determining areas of interest, a hardware-software system for rapid radiological mapping will be developed, applicable in the future for installation in unmanned aerial vehicles, based on dosimetric detectors for measuring ambient dose equivalent, GPS receiver and compact central computer. The maximum area mapping speed will be correlated with the detector sensitivity and its response time. Field measurements will be carried out at at least two research sites where in-situ and laboratory gamma spectrometric measurements will be performed. Radiological measurements will be accompanied by the measurement of meteorological parameters that will enable the development of a correction pilot algorithm for specific meteorological and other physical environmental conditions of nuclear power plants.

Keywords: *Gamma Spectroscopy, Dosimetry, Environmental Monitoring, Radiation Detectors*

S3-163

Techno-economic and Environmental Impact of NPP Krško Lifetime Extension on Croatian Power System Development

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Half of the 696 MW of total capacity of the Nuclear Power Plant Krško (NEK) is used for the Croatian power system, and energy generated for Croatia equaled almost 16.8% of final electricity consumption in Croatia in 2020. After almost 40 years of successful operation, NEK is nearing the lifetime extension for another 20 years for the period 2023-2043. In this paper, we assess the techno-economic and environmental implications of lifetime extension for Croatia. We simulate future developments of the Croatian power system in line with the S1 scenario presented in the national Energy Strategy, and in line with the national Climate-neutral Scenario until 2050. For both scenarios, we assess the implication of cases with and without NEK lifetime extension. Techno-economic implications are evaluated based on the additional investments needed and levelized cost of energy in the power system, while environmental implications are evaluated based on the impacts on emissions of greenhouse gases in the power system. The results point out that the lifetime extension of NEK proves to be economically and environmentally favorable. Moreover, in case of earlier retirement in 2022, the national power system and the economy would be exposed to a bigger risk of market prices and uncertainties (electricity, fuels, emission allowances) in the coming years, as no alternative solution can be built immediately.

Keywords: *nuclear power plant, greenhouse gas emissions, levelized cost of energy, power system, planning, optimization*

S3-164

Power Flows and System Dynamics Influence of NPP Krško Life Time Extension on Croatian Power System

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NPP Krško is, after almost 40 years of successful operation, facing the end of its originally assumed design life. Thanks to a proper maintenance, planning and investment, including replacement and upgrades of old equipment, and upgrades of safety systems, NPP Krško is ready for life extension of additional 20 years. When addressing environmental impacts of any existing energy source it is important to see what are the alternative ways to produce the same amount of energy and what are consequences of operation of both original and potential replacement power sources. In that regard this paper describes the assessment that was performed to explore the influence of NPP Krško on power systems of Slovenia and Croatia in the period between 2023 and 2043. Multi-scenario analysis was performed for assumed development stages of the two power system mainly to assess the importance of NPP Krško in comparison to potentially newly added production, mainly wind and solar. The results are shown from twofold perspectives 1) power flows perspective observing the losses, loadings, voltage conditions and 2) power system dynamics perspective observing the stability of the system.

Keywords: *power flows analysis, power system dynamics*

S3-167

The Approaches to Development of New Build Nuclear Projects in the Czech Republic

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The Czech Republic has ambitious plans for the construction of new nuclear capacities. Nowadays nuclear accounts approximately for the 30% of the electricity generation. At the same time the coal phase-out is already ongoing and the use of a coal in energy mix is planned up to 2038 at maximum. Nuclear energy will become the most significant energy source by 2038 as the new nuclear capacities are planned to be put into operation by 2036. Furthermore by 2038 the country would become net electricity importer. To facilitate all challenges associated with the energy sector transformation and expected lack of the electricity the massive nuclear sources build-up is one of the possible answers to such situation. The Czech Republic is on the verge of the possible nuclear renaissance. In new build program the focus must be put on the proper and complex projects preparation. One of the critical tasks in newly developed project is its adaptation to local conditions, its licensing in the specific country conditions and the economic boundaries set up by all parties involved. Author aims on description of the preparatory phase of the new build projects and the boundary conditions set to ensure that the project is viable in specific Czech environment.

Keywords: *new build, regulatory processes, design adaptation, nuclear planning*

Session 4
Regulatory Practice and Emergency Preparedness (RPEP)

S4-143

Application of a new in situ calibration technique for gamma ray spectrometry and comparison of in situ and laboratory measurements taking into account realistic Croatian conditions

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After a nuclear accident, it is important to quickly measure possible affected area to determine where emergency and remediation measures are needed. In situ gamma ray spectrometry was developed to quickly measure large areas of land following nuclear weapon tests and possible nuclear accidents. However, a proper calibration of detectors for in situ measurements is a long and complicated process. One tool designed to make this calibration quick is the InSiCal software.

InSiCal software can shorten the in-situ calibration procedure to a single measurement (combined with calculations). We decided to investigate if the method can be implemented in Croatia, within expected constraints of emergency situations (especially time constrains).

We compared in situ measurements made with two different HPGe detectors calibrated using the InSiCal software and laboratory measurements of samples collected at the same locations. Detector calibration and in situ measurements were optimized for quickness, simulating time pressure present in case of a nuclear accident.

Measurements of both in-situ detectors were reasonably close – in most cases the confidence intervals overlapped. In-situ measurements generally undershot laboratory measurements. Large uncertainty intervals at energies below 100 keV make short in-situ measurements unsuitable in that energy range. If the range below 100 keV is important, the duration of the measurements must be increased.

Our findings suggest that in situ gamma spectrometry using InSiCal software can provide reasonably accurate data, but some improvements may be needed.

Keywords: *emergency, HPGE spectrometers, InSiCal software, radiation, radionuclide measurements*

S4-144

Methods for Monitoring and Detecting Faults in IoT Dosimetry Instrumentation Based on Machine Learning on Edge Computing Devices

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Due to the importance of safety, reliability and efficiency of dosimetry instrumentation, as well as increasing complexity of the technologies, we are proposing a method for early failure detection that could enable the necessary prompt response. IoT dosimetry sensors are usually required to operate for several years on a single battery and they are often installed in large numbers which place high energy and cost constraints. Therefore, the analysis and prediction itself is increasingly performed on devices that are close to the sensors. The concept of bringing analytical computational resources closer to the sensors themselves is called edge computing. In this work we will consider the application of machine learning for the purpose of fault detection in IoT dosimetry instrumentation as well as the various approaches with which these detections are realized with the help of edge computing devices.

Keywords: *Radiation Dosimetry, Machine Learning, Fault Detection*

^{137}Cs and Naturally Occurring Radionuclides in Soil in Croatia

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Soil contributes significantly to both external and internal exposure to ionising radiation, via direct emission of gamma radiation and soil-to-plant radionuclide transfer, respectively. Next to primordial radionuclides, soil contains radionuclides dispersed during nuclear weapon testing and Chernobyl and Fukushima accidents, especially ^{137}Cs. This motivated us to carry out a systematic research on the radioactivity of soil in Croatia, with a goal to obtain relevant data about the spread of ^{137}Cs in Croatia, as well as about the primordial radionuclides. We had collected samples of the surface layer of uncultivated soil (0-10 cm) at 138 sites from all over the country and measured them for radionuclide activity concentrations by means of high-resolution gamma-ray spectrometry. This resulted in maps of the radioactivity of Croatian soil, containing data on activity concentrations of representative radionuclides in the environment. We focused on ^{137}Cs, and also mapped ^{40}K and radionuclides from ^{232}Th and ^{238}U decay chains. We found that the concentrations of ^{137}Cs tended to increase with altitude, annual precipitation, and vegetation density. Activity concentrations of ^{40}K were the highest in the Pannonian region. The ratio of the concentrations of ^{137}Cs and K in soil, representing the potential for ^{137}Cs entering food chains via uptake by plants, was the lowest in agriculturally important areas in the east of the Pannonian region.

For both ^{232}Th and ^{238}U decay chains, activity concentrations were the highest in the Dinaric region, the lowest in the Pannonian region, and intermediate in the Adriatic region. In particular, relatively high concentrations of ^{226}Ra in the soil of the Dinaric region implied a possibility of an enhanced emanation of its progeny ^{222}Rn into the air. Activity concentrations of ^{210}Pb were additionally elevated in areas with dense vegetation, most probably due to an atmospheric deposition of airborne ^{210}Pb onto the surface of plants and their eventual decomposition on the ground.

We used the obtained results on activity concentrations to calculate the related absorbed dose rate as a measure of external exposure to ionising radiation from soil. The sum of the absorbed dose rates for naturally occurring radionuclides and ^{137}Cs showed that the external exposure was generally the highest in the Dinaric region and the Istrian Peninsula.

Keywords: *gamma radiation; high-resolution gamma-ray spectrometry; radioecology; representative radionuclides; ^{137}Cs, ^{40}K, ^{210}Pb; ^{222}Rn; ^{226}Ra; ^{232}Th; ^{238}U*

S4-155

Regulatory Activities in Preparation for the Project of new Slovenian NPP in Krško

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Some 15 years ago in Slovenia the idea arose of a new nuclear power plant (NPP) construction at a site close to existing plant in Krško. A nuclear renaissance started in several European countries. But in March 2011 the Fukushima accident occurred as a consequence of extreme natural events, a combination of beyond design bases earthquake and tsunami. This accident had a negative impact on the construction of new NPPs in European Union with some countries abandoning nuclear power option while others halted the construction of their plant.

Long-term positive experience with Krško NPP operation as well as significant investment in the plant safety for extreme external events and severe accidents, increased the public confidence in use of nuclear energy in Slovenia. After completion of Krško safety upgrade project at the end of 2021, the time was right to start the new project of JEK2. The first permit obtained was the energy permit already in July 2021 and in January 2022 the documentation for initiative for the national spatial plan was prepared.

The Slovenian Nuclear Safety Administration (SNSA) as the regulatory body responsible for licensing new nuclear facilities actively responded to these activities. A project team was formed that shall prepare the SNSA for such extensive task of licensing the new NPP and will already participate in the process of the national spatial plan preparation as the first stage of licensing process. The cooperation started between the investor GEN Energija and the SNSA as the regulator. The SNSA is also active in international cooperation within OECD/NEA, the IAEA, the WENRA as well as the regulators from countries with bilateral agreements with the SNSA.

The regulatory activities are aimed at increasing the capabilities of the SNSA by employing new personnel and preparing the qualifications criteria as well as training program for the newcomers as well as for the existing staff. The Slovenian legislative framework was also upgraded with new revisions of acts on spatial planning, environmental protection and construction. The SNSA prepares amended regulations with nuclear safety requirements based on WENRA Safety Reference Levels and the IAEA requirements.

Keywords: *SNSA, JEK2, new builds, legislation, siting*

Session 5
Reactor Physics and Nuclear Fuel Cycle (RPNFC)

S5-122

Modelling of Diffusion Neutrons Flux

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The well-known law of attenuation of a narrow particle flow cannot be applied to a diffusion flow. Usually, differences are tried to be taken into account by applying the empirical coefficient of accumulation, but there is no theoretical justification for its value. An appropriate diffusion flow model is required to bridge this gap.

The process of diffusion flux formation in the medium where neutrons diffuse and near the boundary with the medium where they are absorbed is considered. Theoretical calculations are supported by calculations using the Monte Carlo model.

The distribution of neutron concentration nearby the boundary between the diffusion and absorption medium by iterative approximation is presented. The distribution depends on the neutron-physical properties of the diffusion medium. The obtained results are the foundation for further analytical study of the law of attenuation of diffusion flux.

Keywords: *neutron diffusion flux, attenuation, build-up coefficient, Monte Carlo model, neutron concentration distribution*

S5-130

Simulation of ^{252}Cf Neutron Transmission Through an Iron Sphere

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This paper presents an extension of an earlier study to test iron cross-section data from ENDF/B-VII.1 library to be used in Monte Carlo and transport calculations of neutron transmission through an iron sphere. Deficiencies in the iron inelastic data from the previous ENDF/B-V evaluation was a known issue, giving concern for a fast neutron flux underestimation within the reactor pressure vessels. In order to benchmark the next-generation ENDF/B-VI iron data, the U.S. Nuclear Regulatory Commission and the former Czechoslovakian National Research Institute have jointly preformed several experiments in 1990s, addressing neutron leakage spectra obtained for a ^{252}Cf fission source in a centre of an iron sphere. It was shown that the ENDF/B-VI iron cross section, containing several improvements over previous evaluations, will not entirely resolve the neutron spectrum discrepancies observed at high neutron energies. Since safety analyses of reactor pressure vessel embrittlement are often based on neutron transport calculations using specific multigroup cross section libraries, simulation of this benchmark was performed using a hybrid shielding methodology of ADVANTG3.0.3/MCNP6.1.1b codes. Comparison of calculated and referenced dosimeter activation rates are presented for several "standard" nuclear reactions, often used in reactor pressure vessel dosimetry. For that purpose, the new IRDFF-II special library was used as a reference source of dosimetry cross sections. MCNP6.1.1b computed reaction rates were also analysed using previous IRDFF-1.05 special library and general purpose ENDF/B-VII.1 library.

Keywords: *dosimetry, shielding, Monte Carlo, MCNP, ADVANTG*

S5-131

Visualisation of the MCNP-based Mesh Tally File

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This paper presents an updated version of the MTV3D (Mesh Tally Visualization in 3D) program with Windows graphical user interface for visualization of the MCNP-based mesh tally objects. Several improvements over the previous version are addressing better figure export functionality ("Save as" option), switching between linear and logarithmic values on axes, dynamic figure scaling in active window, inversion of relative errors from max to min values, etc. MCNP is a well known and widely used general purpose Monte Carlo computer code for neutron, photon and electron transport simulation through arbitrary three-dimensional configurations. An important feature of the code is a graphical display of the simulation model using auxiliary program, such as X-window server, which is useful for geometry error-checking during model setup and visualisation of Monte Carlo results from a mesh tally file (i.e. meshtal file) over a structured *xyz* mesh. Such inspection of the model is useful for the end user, providing an insight of the Monte Carlo convergence process in a phase space and effectiveness of the selected variance reduction parameters in shielding calculations. Basic features and functionalities of the updated MTV3D version are presented on some selected hybrid-shielding problems involving ADVANTG3.0.3/MCNP6.1.1b codes.

Keywords: *mesh tally, hybrid shielding, Monte Carlo, MCNP, ADVANTG*

S5-132

Characterization of the GBC-32 Fuel Assembly Source Terms

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This paper presents burnup/depletion calculations of the typical Westinghouse 17x17 fuel assembly to be used as a radioactive waste package in a Generic Burnup Credit cask with 32 elements (GBC-32). This first phase is addressing spent fuel source terms calculation while evaluation of the shielding performance of the GBC-32 cask is planned for the second phase. The TRITON-NEWT methodology of the SCALE6.1.3 program package was used in a tandem with ORIGEN-S code for deterministic 2D calculation of the GBC-32 fuel assembly neutron multiplication factor, providing spatial-temporal fluxes and isotopic concentration change. The burnup simulation was done up to 60 GWd/MTU with sensitivity analysis of relevant physical parameters influenced by the working cross section library. This approach also allowed generation of the specific user-defined collapsed cross section libraries as a function of fuel enrichment and burnup level. Calculation of isotopic concentrations, decay heat, neutron-gamma spectra and major actinides activity for different fuel assembly cooling periods was performed using ORIGEN-ARP module.

Keywords: *SCALE, NEWT, TRITON, burnup, depletion*

S5-137

Spent Fuel Characterization to Support NPP Krško Spent Fuel Dry Storage Project

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Currently, all spent fuel in the NPP Krško is stored in the spent fuel pool located in the fuel handling building of the plant. After the Fukushima accident in 2011, it was decided to relocate some of the fuel in the dry storage casks to increase safety and storage capability for spent fuel assemblies. Dry storage concept is safer and more reliable storage solution, compared to spent fuel pool storage concepts, since it uses passive design systems. Within NPP Krško Spent Fuel Dry Storage (SFDS) project activities, the construction of a dry storage building started in March 2021 after a construction permit was issued. The building will host Holtec International HI-STORM FW casks consisting of a steel canister inserted in a concrete overpack. The relocation of fuel assemblies from the pool to the new storage facility will start with the relocation of about 600 fuel assemblies in 16 casks. The second phase, scheduled to begin in 2028, will fill another 16 casks. The final phase will be completed five years after the power plant is scheduled to shut down in 2048. The storage facility will be built to last 100 years, after which permanent disposal solution will be needed.

The main function of the storage casks is to provide sufficient spent fuel cooling and shielding against gamma and neutron radiation emitted by the radioactive contents. The adequate characterization of the spent fuel, where the decay heat, fuel activity, photon and neutron source term are determined, is an important parameter as it affects the loading of the casks. The loading constraints arise from the safety and design requirements for the specific storage casks.

Jožef Stefan Institute (JSI) is supporting NPP Krško SFDS project. In the first phase fuel assembly database was verified by comparison and harmonization of the plant data with the independent JSI database. These parameters, such as fuel enrichment, burnup, fuel operational conditions during irradiation, number of BPR and IFBA rods, dictates spent fuel behaviour predictions and are needed in the process of the spent fuel characterization. Input models for the stochastic neutron transport code Serpent2 and deterministic codes from the SCALE package has been developed. Comparison of the spent fuel decay heat, activity and neutral particle emissions for cooling times up to 50 years has shown good agreement between both code systems. In the final project phase, NPP Krško spent fuel sensitivity study has been accomplished based on the developed TRITON/NEWT model and spent fuel predetermined parameters range. The study identified most influential fuel parameters needed in the development of the optimal calculation model that will be used in the routine calculation of observables.

Keywords: *Spent nuclear fuel, decay heat, activity, photon source term, neutron source term, sensitivity analysis, spent fuel dry storage*

S5-141

Reactor Physics and Thermal Hydraulics Analyses for the OECD/NEA MPCMIV Benchmark

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In order to complete the Multi physics Pellet Cladding Mechanical Interaction Validation (MPCMIV) benchmark technical specifications, reactor physic and thermal hydraulic analyses have been performed. The work presented in this paper aims to evaluate some of the missing Boundary and Initial Conditions necessary to complete the technical specifications, and also to perform some of the benchmark exercises connected with thermal hydraulic simulations. A far as the thermal hydraulic area is concerned, the analysis is carried out with the RELAP5 code. It is focused on the modelling of the in pile loop 1 located inside the R2 reactor core, in which a test fuel rodlet is inserted to perform some power ramp tests. The activity consists in the development of the simulation model of the in pile tube, the demonstration of the steady state achievement and the transient analysis of the first selected test, validating the simulation results against the benchmark experimental data. Considering the reactor physic area, the Monte Carlo code Serpent 2 is used to perform some single assemblies burn up calculations. The aim is to evaluate the initial composition of the fuel assemblies loaded in the core loadings of interest of the benchmark. Moreover, the temperature values to be used in the Serpent simulations are derived with thermal hydraulic simulations of the single assemblies. Further developments of the work will include the full core cycle analysis to validate the isotopic compositions and the complete model of the main circuit, using the gamma heating from the reactor physics calculations. Finally the TRANSURANUS fuel performance code will be adopted to compare the results against the available experimental data. A multi-physics effort is required to carry out the MPCMIV benchmark and appropriate coupling approach will be investigated and tested against the benchmark experimental results.

Keywords: *MPCMIV, Multi physics, RELAP5, Serpent, Validation*

S5-153

Applicability of Legacy and New Calculation Tools for 2D Fuel Assembly Depletion as defined by VERA Benchmark

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Consortium for Advanced Simulation of LWRs (CASL) published in year 2015 'Specification for the VERA Depletion Benchmark Suite'. The benchmark contains specification for 16 fuel assembly configurations and 10 pin cells configurations covering almost all features of modern LWR fuel. We wanted to see how, so called, legacy 2D spectral codes perform against more modern deterministic and probabilistic codes. In calculation, we used 3 deterministic codes: our FA2D code, Triton and Polaris from SCALE 6.2.4 package and one probabilistic Serpent 2.1.32 code. In addition, initial effective multiplication factor (without depletion), for each configuration, was calculated with MCNP6.2 code. The influence of different cross section libraries was studied for codes where more libraries were available for calculation. We were interested in differences in calculated effective multiplication factors, group constants and pin power factors and required CPU time and memory allocation.

Keywords: *2D spectral codes, VERA benchmark, group constants, pin powers*

S5-159

Preliminary Assessment of Parallel Efficiency of SCALE CSAS6 and T6-DEPL Sequences

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Reactor physics analyses of complex nuclear systems simulations aiming to achieve accuracy rely to a large degree on Monte Carlo methods. The challenge then becomes reducing the statistical uncertainty which generally requires simulating a large number of neutron histories. Achieving required statistical convergence within acceptable turnaround time requires in most cases parallel simulations.

The SCALE code suite developed and maintained by Oak Ridge National Laboratory (ORNL) is presently composed of 11 end user products with capabilities ranging from radiation shielding to sensitivity and similarity analysis. Of particular interest for reactor design is KENO-VI continuous energy (and multigroup) Monte Carlo radiation transport code with corresponding sequences for criticality and depletion evaluation, CSAS6 and T6-DEPL. SCALE6.2 contains several modules and sequences that have distributed memory (MPI) parallelism, including KENO-VI. SCALE6.2.4 with these parallel capabilities has been successfully built and installed on the Georgia Tech PACE cluster.

The paper reports results of preliminary testing of parallel performance for several representative problems, from simple to more complex ones, from static no depletion to depletion cases, evaluated for weak and strong scaling, on a single multi-CPU node as well as on multiple nodes. The simple Godiva problem achieves a maximum strong scaling speedup of about 7, and this does not improve significantly for the weak scaling, suggesting that the inherent bottleneck is in very limited computational effort required per particle history. SCALE parallel diagnostics provides useful data and supports this conclusion. The uranyl nitrate solution test problem provides better parallel performance due to longer neutron histories, but is still too simple to significantly benefit. More complex MSR depletion problem (T6-DEPL sequence) achieves around 80% parallel efficiency on 24-96 CPUs. Finally, a more relevant problem representing a complex FHR fuel assembly geometry achieves parallel efficiency exceeding 90% on single node and multiple nodes, tested on up to 96 CPUs. Used cautiously, these findings can provide a useful a priori indication of possible speedup and a guidance how to improve it.

Keywords: *Monte Carlo, SCALE, CSAS6, parallel efficiency*

S5-170

The French Reprocessing Solution and Its Recent Evolution

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Used fuel reprocessing offers several benefits for operators: a) reduction of risks and costs, b) enhancing public support, c) allowing safe and long-term storage of final waste, d) rationalization of geological repository.

In France, a set of decrees have been recently issued by the government allowing the return of the global inventory of Universal Canister of Compacted Waste (UCC) and Universal Canister of Vitrified Waste (UCV) from two countries of origin using different ratio of UCV and UCC for each country, provided that all metallic mass and radioactive activity of both countries leaves France.

The presentation will focus on this new approach integrating the flexibility in the system of attribution for UCC and UCV providing additional value to reprocessing clients.

Keywords: *Reprocessing, used nuclear fuel, universal canister of compacted waste, universal canister of vitrified waste*

Session 6
Severe Accident Analyses and Risk Assessment (SAA)

S6-107

Decision Support Tool for Severe Accident Management

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The project NARSIS – New Approach to Reactor Safety ImprovementS – is making scientific steps towards addressing the update of some elements required for the safety assessment of nuclear power plants. These improvements mainly concern:

- Natural hazards characterization, in particular by considering co-current external events, either simultaneous-yet-independent hazards or cascading events, and the correlation in intra-event intensity parameters.
- Vulnerability of the elements to complex aggressions, with the integration of new approaches such as vector-based fragility surfaces and reduced models.
- Better treatment of uncertainties through adoption of probabilistic framework for vulnerability curves and non-probabilistic approach to constraining the "expert judgments".
- Development of decision support tool for severe accident management.

The decision support tool for severe accidents – called Severa – has been developed in this project. It is a prototype demonstration-level decision support system aimed at supporting the technical support center (TSC) while managing a severe accident. Severa represents, stores and monitors selected physical measurements of the NPP. It assesses the current state of barriers: core, reactor coolant system, reactor pressure vessel and containment. The prediction of future accident progression, if no action is undertaken, is one of basic functions. The support tool provides a list of possible management recovery strategies and courses of action. The applicability and feasibility of possible actions in the given situation is identified. For each action, Severa assesses possible consequences in terms of probability of the last barrier (containment) failure and estimated time window for failure. At the end, Severa evaluates and ranks the feasible actions, providing recommendations for the TSC. The verification and validation of Severa is performed in the project and is described in this paper.

Keywords: *Severe Accident Management, Decision Support System, Decision Model*

S6-112

MAAP 4.07 Analysis of Long Term Containment Heat Removal After Reactor Vessel Failure (DEC-B) for Nuclear Power Plant Krško (NEK)

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This paper presents the MAAP 4.07 analysis of containment heat removal after reactor vessel failure resulting from the initial Station Blackout (SBO) accident. The accident is analyzed taking into account mitigation measures for heat removal from the containment using alternative equipment (Alternative Residual Heat Removal (ARHR) pump and heat exchanger (ARHX) and, also, Alternative Safety Injection (ASI) pump). The mitigation actions are taken according to NEK Severe Accident Mitigation Guidelines (SAMG).

There are several possibilities to remove the heat from the containment once the reactor vessel fails and, for all of them, the necessary condition is to have the sufficient source of water (Residual Water Storage Tank (RWST), Alternative Boron Water Tank (ABWT) or other) and the appropriate heat exchanger available. Two options are presented within this paper:

Injection to the Reactor Coolant System (RCS) using ASI pump and recirculation (sump to RCS) through ARHR system via ARHX,

Spraying the containment through Containment Spray (CI) system using ARHR pump and, then, recirculation (sump to spray) through CI and ARHR systems via ARHX.

The results show that the containment heat removal can be done with either of analysed ways if the water is provided for recirculation (assumed containment level 3.9 m ~ 760 m^3). However, with the fact that the reactor cavity is not flooded, the cooling using ASI will initially result in significant containment pressure increase because the water is spilled through the RCS over the hot molten core debris. Therefore, it must be stated that the preferable way of containment pressure reduction, once the vessel has failed, is by using the containment spray. On the other hand, if RWST is not available, then the initial water delivery cannot be made from ABWT via CI system because these options are not foreseen. It shall also be pointed out that, if the active containment heat removal is started early enough, the PCFVS opening would be prevented and no fission products shall be released to environment.

Keywords: *station blackout (SBO), containment heat removal, design extension conditions (DEC), MAAP 4.07, Nuclear Power Plant Krško (NEK)*

S6-129

The Large Minimal Cut Sets Compact Representation with Binary Decision Diagrams

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Since the beginning of application of the fault tree analysis in nuclear energetics it has been obvious, that a more accurate insight into the reliability of the observed system relies on the understanding of the complete or, at least, the most significant parts of sets of minimal failure components (MCSs). However, the determination of MCSs, even with simplest systems (or subsystems) modelled by a coherent fault tree, turns out complex, facing at least two basic problems. The first, being the time complexity of algorithms employed for the determination of the complete or partial set of MCSs, while the latter problem relates to a space complexity of the same sets recording. More recently, binary decision diagrams (BDDs) have been developed, enabling indirect recording of fault trees by applying indicator variables for the component failure state within the system. This state is habitually presented by means of the Bernoulli random variable for basic, component associated events, i.e., in the way that the particular component is associated with the probability of the occurrence of a failure event. Such association establishes a link between the logical function presented by a coherent fault tree model and the probabilistic model for analysis (PRA). Now, a qualitative and quantitative analysis on a fault tree model may be carried out with BDDs by applying known algorithms for the determination of minimal disjunctive normal form of the logical function presented by the coherent fault tree, which represents the logical recording of a set of minimal cuts. Thus, not only do BDDs show (under the condition of an appropriate variable order) an acceptable time complexity for the implementation of algorithms for determining MCSs but also enable a compact recording of complete or partial sets of MCSs singled out in that way. The exceptional compactness of minimal cut set recordings gained by the BDDs technique, makes it possible to ensure, the recording of complete set of MCSs. The accessibility of the complete set of MCSs by means of BDDs allows accurate calculations from a probabilistic model. The results indicate compactness of recordings (mostly 10^15 MCSs or more are being recorded), hence, this technique facilitates the recording of MCSs for fault tree models of the most complex events such as core damage, which have been described typically with more than 1000 basic events.

Keywords: *Probabilistic Risk Assessment (PRA), Fault Tree Analysis (FTA), Binary Decision Diagrams (BDD), Minimal Cut Sets (MCS)*

S6-133

Baseline ASYST Calculations – Estimates of the Likely Reactor Behavior in the event of an SBO-related Event for Ukrainian VVER-1000 NPPs

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After the Russian invasion of Ukraine in February 2022 and attack on their nuclear power plants (NPPs), Innovative Systems Software (ISS), contacted the Energy Safety Group LLC (ESG) in Ukraine, and offered their support in the event that any accident management strategies might need to be updated in light of the increased risk of station blackout (SBO) conditions. ESG had recently published the results of their "Post-Fukushima" safety analysis of the Zaporizhzhya NPP Unit 1 using a combination of RELAP5 and MELCOR. ISS, and other collaborators (and co-authors) had provided similar support for the emergency assessment of the Fukushima Daiichi reactor behavior using RELAP/SCDAPSIM. This support was initially provided to the International Atomic Energy Agency (IAEA) emergency response team during the Fukushima accident, and later to the Japan Atomic Energy Agency (JAEA), in support of their post-accident decommissioning research activities. Other collaborators and co-authors have performed a wide range of VVER-1000 safety assessments using RELAP/SCDAPSIM and, more recently, the new best estimate integral system thermal-hydraulic code, ASYST. Fortunately, much like the Fukushima Daiichi support activities where the Comision Nacional de Seguridad Nuclear y Salvaguardias (CNSNS), the Mexican Nuclear Regulatory Authority had provided access to their detailed RELAP/SCDAPSIM Laguna Verde BWR input models, ISS, and their VVER-1000 collaborators, have representative RELAP/SCDAPSIM VVER-1000 input models, derived from models originally provided by the Institute for Nuclear Research and Nuclear Energy (INRNE) in Bulgaria.

In preparation for any analysis that might have been required on an accelerated time frame, it was decided to perform a series of baseline calculations that could then be used to provide preliminary assessments of the reactor and containment behavior in the event of an SBO with the added possibility of externally caused damage to the containment, reactor coolant system (RCS), or associated safety systems. This approach is similar to that used in support of the IAEA emergency response team as

the Fukushima Daiichi accident was in progress. It was decided that ASYST would be used for the analysis to allow direct comparisons with the original ESG calculations using RELAP5 and MELCOR. This paper describes the basic features of ASYST, the basic features of VVER-1000 input models that were used, and a summary of some of the key results from these baseline calculations.

Keywords: *RELAP/SCDAPSIM, ASYST VER3, SBO, VVER-1000, Baseline studies*

S6-136

Safety analysis of SBO in BDBA scenarios for the three loop Westinghouse NPP using PCTran simulator in comparison with TRACE results

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This paper presents the study results of a station blackout (sbo) scenario causing by the complete loss of electrical power on the three loop westinghouse nuclear power plant. Since many safety systems for reactor core residual heat removal and containment heat removal rely on ac power. The prolonged station blackout might eventually lead to a severe accident with significant core degradation. In this scenario, the turbine driven auxiliary feedwater pump is assumed to be unavailable as well.

A few seconds after the initiation of sbo, the reactor trip signal is generated due to low reactor coolant flow rate. At first, the reactor cooling system pressure increases since the reactor cooling pump trip causes an increase in reactor coolant temperature and the water level slightly drops due to the reactor trip. In the secondary side of steam generator, since feedwater and steam flow do not exist, evaporated sg water pressurizes the sgs and is slowly discharged through the mssvs (main steam safety valves).

The pctran simulator version 6.0.4 based on a generic conventional three-loop pressurized water reactor (pwr) has been used to evaluate this accident.

As a result, reactor cooling system liquid volume decreases, and the reactor core is uncovered at a long time causing the zirconium oxidation and hydrogen generation.

These results are compared with similar calculation using trace system code.

Keywords: *PCTRAN simulator, SBO, PWR, BDBA, TRACE*

S6-151

Thermal Response of External Duct Piping in Case PCFV Actuation During Prolonged NEK SBO

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NPP Krško MELCOR input deck with explicit model of PCFV system was used for calculation of prolonged SBO with active and passive PCFV actuation. The main focus of analysis is evaluation of thermal hydraulic conditions (pressure, temperature, relative humidity, H2 and noncondensible gases fractions) in external PCFV duct piping and calculation of external temperature of metal duct wall.

Keywords: *MELCOR, NEK SBO, PCFV, external duct temperature*

S6-156

MelSUA – an Open-Source MATLAB Toolbox for Sensitivity and Uncertainty Analysis with MELCOR Code

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The MelSUA (Melcor Sensitivity and Uncertainty Analysis) is an open-source tool dedicated to sensitivity and uncertainty (S&U) analysis with MELCOR computer code. The tool was implemented as a toolbox for the MATLAB computing environment.

This paper presents the methodology, development process, implementation, and code structure. Additionally, basic usage, guidelines, and example input and example results are presented. The basic aim of the software is to support MELCOR analysts familiar with MATLAB in performing sensitivity and uncertainty analysis for severe accident studies in Nuclear Power Plants. The main advantage of the MelSUA is the use of a popular MATLAB environment with powerful statistical toolboxes, which allows complex data analysis in an easy manner with matrix-based mathematical functionality and advanced post-processing. At the current stage of development, the tool allows performing Monte Carlo or Wilks-based uncertainty analysis using Simple Random Sampling or Latin Hyper Cube sampling for parameters with various continuous and discrete probability distributions. The tool is equipped with routines for sensitivity analysis with linear regression and computation of popular simple correlations like Spearman or Pearson. It works with simple XML or MAT input files, automatically processes input decks, and generates ready-to-run execution scripts for Microsoft PowerShell.

Currently, the beta version of the tool is available on the GitLab repository (www.gitlab.com/darczu/x-core/-/tree/master/Modules/MLC_package/SenstivityUncertanityAnalysis). The tool was developed as a contribution to the NARSIS (New Approach to Reactor Safety Improvements) Horizon 2020 research project.

Keywords: *MELCOR, open-source, sensitivity and uncertainty, severe accident, Matlab*

S6-158

Importance Ranking of Diverse Safety Issues for Operating NPPs

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Any operating nuclear power plant (NPP), as a facility with potential for radioactive release, is subject to numerous safety reviews with different purposes and objectives. Some of the safety reviews are, by their nature, general and extensive in terms of different safety factors or safety attributes which are covered. An example of such a review is a Periodic Safety Review (PSR) which is promoted by the International Atomic Energy Agency (IAEA) and a number of national safety authorities. PSR is many times used as a means for verifying whether a plant which has been operated in long term (e.g. over decades) is still as safe as originally intended and particularly in the context of new safety standards which have come into place after the time of plant's initial operation. The other reviews may, depending on the objective, be targeted at particular safety factor (e.g. ageing management or equipment qualification or safety analyses). The reviews may be initiated and implemented by various stakeholders, including utilities, industry and regulators. Both of the mentioned cases (single general review or multiple targeted reviews over a time period) can generate an inventory of safety issues which need to be addressed but may be very different in their nature and implications, as well as in benefits or resources associated with their resolutions. While some of the issues may be directly related to operational safety (e.g. non-compliance with single failure criterion or aging-related degradation of safety features), for some others the link to operational safety may not be explicit (e.g. comparison of safety bases against the newly emerging methodologies or issues observed with regard to so called "soft factors").

The paper will discuss types of safety issues which may emerge from general or targeted safety reviews and outline some basic principles for comparison and importance ranking of diverse issues, as needed many times in order to develop an action plan for keeping or improving the plant safety level.

Keywords: *operating NPP, safety review, safety issues, ranking*

S6-161

MELCOR-TO-MELCOR Coupling Method in Severe Accident Analysis Involving Core and Spent Fuel Pool

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In Severe Accidents (SA) Management, a lot of effort has been spent to prevent the occurrence of core damage constituting the begin of the SA phase in the accident progression. However, the core in the vessel is not the only source of fission products in the containment as the Spent Fuel Pool (SFP) hosting the fuel removed by the core is, in some NPP, inside the containment and SA conditions can occurs in the pool. This is especially important in reactors having proximity between the RPV and SFP such as the VVER-1200. This close proximity implies that any SA occurring in the SFP potentially affects the RPV and vice-versa. This potential combination might cause unexpected evolution in the SA progression to whom the safety systems are not able to contain.

MELCOR code is a widely used, flexible powerful SA code but it is incapable (due to the uniqueness of the COR package use inside the same input) to reproduce a situation in which both the fuel in vessel core and the fuel in the SFP, inside the same containment, are going to experience a severe accident scenario.

The current study presents a MELCOR-to-MELCOR coupling method to simulate simultaneously scenarios with both, core and SFP, as active regions capable of H2 generation, fuel damage and FP release in a VVER-1200 NPP. The coupling is performed by running two simulations in parallel and with the data exchange supervised and managed by a dedicated Python coupling supervising script developed at NINE.

Keywords: *MELCOR, SEVERE ACCIDENT, SPENT FUEL POOL, COUPLING, VVER-1200*

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

S7-108

National concept for decommissioning and radioactive waste management

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The Czech Republic operates six nuclear power reactors (VVER), four VVER-440/213 reactor units at Dukovany and two VVER-1000/320 reactors at Temelín. Czech government is fully responsible for the decommissioning and for the disposal of all radioactive waste (RAW) including spent nuclear fuel (SNF). The decommissioning strategy and concept of RAW management and SNF management in the Czech Republic which set out the relevant principles, objectives and procedures to be followed was approved by the Czech government. The legal framework is set by Atomic Act and subsequent legal regulations.

The Concept for decommissioning and radioactive waste is based on the current situation concerning low-level and intermediate-level RAW management, the development of a deep geological repository for RAW and SNF disposal, legislative changes, Government programming documents and international experience and trends. The Concept is further motivated by preparations for the construction of a new nuclear unit(s) in the Czech Republic, legislative developments within the EU and IAEA and OECD/NEA recommendations.

Actual decommissioning plans for Czech nuclear power plants includes both decommissioning strategies, immediate and deferred decommissioning. Deferred decommissioning involves a 30 years safe enclosure period following SNF removal and facility preparation and modification.

The purpose of the contribution is to share the results of ongoing decommissioning planning and implementation activities for preparation of decommissioning of Czech Republic NPPs. The contribution will focus in particular on the following:

- Approach for decommissioning of NPPs in the Czech Republic.
- The strategy of the long-term back-end nuclear fuel cycle in the Czech Republic.
- Time frames.
- Spent fuel storages.
- Radioactive waste management.

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

S7-115

KBS-3V and axial canister emplacement of SNF – comparison of disposal concepts

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Although KBS-3V is probably the most thoroughly analysed spent nuclear fuel (SNF) disposal concept, and it will be the first concept applied in an actual repository (Posiva, 2022), it is debateable whether KBS-3V is the best and the most feasible concept. Disposal tunnels will be constructed by the drilling and blasting method and the canister emplacement holes by large diameter drills, which means a greater excavation disturbed zone (EDZ), more excavated material and potentially complicated stress/strain models. Another issue is the complicated separate ventilation system and potential problems with bentonite buffer emplacement.

An alternative concept, axial emplacement in disposal tunnels, developed by Nagra and NWMO, offers a smaller repository footprint (for the same number of SNF canisters), a significantly reduced EDZ, a reduced volume of excavated material, a simpler ventilation system and emplacement of the bentonite buffer in the form of pellets.

In this paper, both aforementioned concepts will be compared and discussed from the mining engineering point of view.

References

Posiva (2022) Safety case for the operating licence application: Performance Assessment and Formulation of Scenarios (PAFOS). Posiva Report 2021-06, Posiva, Eurajoki, Finland.

Keywords: *KBS-3V, axial disposal, concept, SNF, repository*

S7-117

Use of natural analogues to support the safety case for a radioactive waste repository

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Despite the earliest mention of the use of natural systems information in the assessment of repository engineered barrier system behaviour going back to 1955 and a seminal conference held in Princeton on radioactive waste disposal by the US Atomic Energy Commission (USNRC 1957), natural analogue (NA) information is often overlooked in safety cases (SC) for radioactive waste repositories. Emphasis is normally placed on laboratory, underground research laboratory (URL) and modelling output and, in part, this is justified as, historically, much NA data have been little more than qualitative (see the discussion in Alexander & Reijonen, 2022). However, a focussed national NA programme (see Alexander et al., 2015; Posiva, 2021, for examples) can produce quantitative data of direct relevance to the long-term SC which complements the shorter-term information produced in the standard laboratory/URL/modelling approach (Figure 1). Here, an approach (Reijonen & Alexander, 2022) to develop a structured, strategically-driven NA programme to support the national repository programme will be presented and discussed.

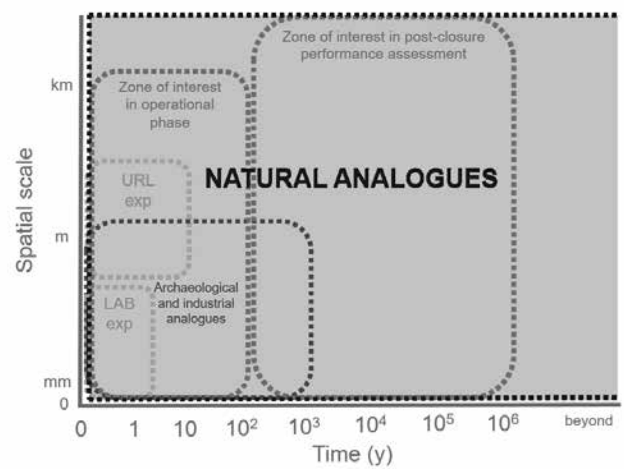

Figure 1. Schematic illustration of the spatial and temporal scales for experiments and NAs data and the scope of assessing repository performance (Reijonen & Alexander, 2022)

References

Alexander, W.R. & Reijonen, H.M. (2022). Natural analogues – a catalogue. GTK (Geological Survey of Finland) Research Report for NWS (Nuclear Waste Services) Ltd (UK). GTK, Espoo, Finland (*in press*).

Alexander, W.R., Reijonen, H.M., McKinley, I.G. (2015). Natural analogues: studies of geological processes relevant to radioactive waste disposal in deep geological repositories. Swiss Journal of Geosciences 108, 75-100. DOI 10.1007/s00015-015-0187-y

Posiva (2021). Safety Case for the Operating Licence Application – Complementary Considerations (CC). Posiva Report 2021-02. Posiva, Eurajoki, Finland.

Reijonen, H.M. & Alexander, W.R. (2022). Natural analogues – strategy for implementation for the NWS programme of geological disposal. GTK (Geological Survey of Finland) Research Report for NWS (Nuclear Waste Services) Ltd (UK). GTK, Espoo, Finland (*in press*).

Keywords: *natural analogues, safety case, radioactive waste, repository, underground research laboratory*

S7-145

New Laboratories for Radiation Protection and Radioecology @ IMROH

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Institute for Medical Research and Occupational Health (IMROH) is at the end of multiyear project: Research and Educational Centre of Environmental Health and Radiation Protection – Reconstruction and Expansion of the IMROH. At the end of the project, the Institute will be expanded with a new building of 6,785.15 m2, while its existing building of 2,067.41 m2 will be completely renovated. Radiation Protection Unit at the Institute specializes in radioecology and radiation protection, and it is continuously engaged in research of radioactive contamination of the environment by natural and fission radionuclides. This contribution presents new laboratories and current capabilities of Radiation Protection Unit at the Institute. Our new laboratories include three new coaxial HPGe gamma spectrometry systems with relative efficiencies up to 130%, electrically cooled in-situ HPGe detector, alpha and beta radiation counting systems, radon detectors, in-situ measurement instrumentation, H*(10) dosimeters for continuous monitoring, sample preparation systems and modeling simulation software solutions.

Keywords: *New Facilities, Instrumentation and methods*

S7-146

New Laboratories for Radiation Dosimetry and Medical Physics @ IMROH

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The Institute for Medical Research and Occupational Health (IMROH) is a multidisciplinary scientific institution with more than 70 years of experience in researching the mechanisms of action of various harmful chemical and physical factors and lifestyles on health and the environment. The Institute has a leading role in Croatia in research in the numerous fields that include dosimetry and protection against ionizing radiation, radiobiology, radiocontamination in the environment, as well as human exposure to these contaminants. Institute is at the end of multiyear project: Research and Educational Centre of Environmental Health and Radiation Protection – Reconstruction and Expansion of the IMROH. The purpose of this project is to increase and improve the current IMROH infrastructure and its research equipment, and with the accompanying organizational reform to establish a Research and Education Center for Health and Medical Ecology and Radiation Protection. Building appropriate spacious infrastructure and investing in modern scientific equipment will significantly increase the scientific excellence and visibility of IMROH in the field of existing research. This contribution gives in depth presentation of new laboratories and current capabilities of Radiation Dosimetry and Radiobiology Unit at the Institute. Unit has always played important role in Croatian scientific and professional areas dealing with radiation dosimetry, radiation protection and medical physics. This role will become strengthened with new laboratories that focus on novel technological developments in instrumentation and measurement methods as well as through acquisition of cutting-edge personal dosimetry systems and in-situ dosimetric instrumentation.

Keywords: *New Facilities, Instrumentation and methods*

S7-152

Thermal Model of HI STORM FW Cask for COBRA-SFS Code

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HI STORM FW cask as used in NEK SFDS project was modeled using COBRA-SFS code. Spent fuel decay heat is calculated using ORIGEN-S for representative cask loaded in NEK Campaign 1. Steady state calculation was performed for typical environmental conditions within NEK Dry storage building. Sensitivity calculations were used to analyze influence of assumptions made in model development.

Keywords: *HI STORM FW cask, COBRA-SFS, thermal model*

S7-157

Gamma Waste Assay System Upgrade

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Krško NPP struggles with a lack of storage space in dedicated LILW storage. Beside stored radwaste packages, some dedicated equipment including two waste assay systems used to occupy a lot of space in the storage building.

After a new building for radwaste operations was constructed, the equipment from the storage building was moved out. The two waste assay systems were reconstructed and functionally combined into one system being capable of measurement of all geometry packages from up to date plant operation. A close cooperation with equipment supplier on waste assay system reconstruction resulted in both, space saving as well as cost effective solution.

Operational LILW forms are result of plant waste streams, typical for pressurised water reactors. However, conditioning and treatment is dominantly oriented in as low as reasonably achievable storage volume. This is due to very limited available capacity of onsite interim LILW storage. A lot of attention is put on characterisation of waste packages on hard gamma emitters. Gamma spectrometry measurements have to be adopted to properties of big variety of waste packages. Waste assay system has to be capable of covering measurements of packages with large span of mass, density, homogeneity and activity. Some packaging for higher activity of waste form is manufactured with a thicker wall to provide shielding during the manipulation early in the process.

Single, custom designed waste assay system is in use at the plant, which has the capability of measurements with transmission source suitable for assay of lower activity non-homogeneous packages. On the other hand, there are mathematically modelled geometries developed for higher activity homogeneous and shielded packages.

Radiological properties of the waste are essential part of data package to be reported to the state central database as well as future data package to be handed over at waste transfer for disposal or long-term storage.

The Krško NPP started operation in 1981 with an onsite storage for LILW, designed as a 5-year interim storage.

Operation with practically no free storage capacity in LILW is bottom line non-conservative. There is a very limited capability for managing waste from mitigation of unplanned events. Dose to operating staff is rather high. Waste characterisation is crucial for waste hand over for disposal or long-term storage.

Keywords: *LILW, gamma waste assay, characterisation, waste storage*

S7-160

Radioactive Waste Status – World Regions and European Union Inventory

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Information about radioactive waste (RW) management including inventory is available from multiple sources (e.g., IAEA and European Commission). The IAEA is collecting voluntary national profiles under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The European Commission is requesting reports from Member States in accordance with Council Directive 2011/70/Euratom establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste. Every three years findings from these reporting is published by the IAEA and EC. Both sources provide comparable status and trends related to RW policies, frameworks, and programs. Significant focus is related to waste and spent fuel inventories including developed practices and technologies.

This paper is reviewing latest available reports from the Commission (including Commission staff working documents) and the IAEA with focus on the RW inventory and status of its disposal. Paper is presenting some findings related to overall situation in the European Union (EU) and inventory comparison between EU member states including some comparison with the reporting from the IAEA related to the whole world. The presentation of radioactive waste and spent fuel inventories is made with values normalized per person and land area. This was done with intention to improve understanding of the scale of the problem related to RW management. Relative scale of the problem is also illustrated by comparison with inventories for hazardous waste.

There is 264000 t of spent fuel worldwide, which this can be expressed as 2 g/km2 and 0.04 g/capita. Total amount of all categories of radioactive waste is 37.6 million of m3, and this can be also expressed as 290 l/km2 and 5 l/capita. Majority of RW (92%) is very low or low level and 81% is already disposed of. In comparison every year about ten times more of hazardous waste is created worldwide (~50 kg/capita/y). These numbers illustrate that amounts of RW are both absolutely and relatively not so big. Status reports with the high percentage of RW disposed of are showing that RW is routinely manageable. This is also including the management of high level RW considering that Finland is very soon opening permanent disposal with several other countries following.

Keywords: *radioactive waste inventory, waste directive, hazardous waste, disposal*

S7-165

Discussion on deep borehole disposal of spent nuclear fuel

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Deep borehole disposal (DBD) of spent nuclear fuel (SNF) and/or high-level radioactive waste (HLW) is, at the moment, a much-debated topic. Some experts think that DBD is a potentially workable solution for the disposal of SNF and/or HLW, especially for small programs, and some think that it is insufficiently researched and technologically unfeasible concept for which disposal containers have yet to be designed and technology tested in life-size. Although the research process for this concept lasts for more than 50 years and has intensified recently, confirmation that the concept is feasible has not yet been achieved.

The original DBD concept is not new, however, it is still one of the least developed and researched concepts. Although, the petroleum industry has an abundance of experience in theoretically similar technology, disposal of SNF and HLW is a much different problem than drilling and production of oil and gas. Thus, it is inappropriate to draw a direct parallel between the technology of drilling the wells for oil and gas production with the technology of making boreholes for SNF and HLW disposal.

The main reason for favoring this concept is the idea of disposal at depths of more than three kilometers, with minimal rock destruction and without creating underground facilities resembling mines. Several problems are still not solved: achieving the appropriate borehole diameter at the required depth; preventing canister jams, canister design, canister installation and borehole-closing technology.

Keywords: *deep borehole disposal, spent nuclear fuel, high level waste, concept, repository.*

S7-166

The Need for Additional Processing of Krško NPP Radioactive Waste

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For the purpose of radioactive waste management in Croatia, the Centre for Radioactive Waste Management will be established at Čerkezovac site. The half quantity of low and intermediate level waste (LILW) from Krško NPP will be conditioned into reinforced concrete containers and then stored in a long- term-storage facility following with disposal in near surface vault type repository.

As a part of the Centre design development the preliminary Waste Acceptance Criteria (WAC) for storage/disposal was proposed. The preliminary WAC has been developed in accordance with the existing practice in the field of near surface disposal. However, the fact remains that preliminary WAC have yet to be approved by the Regulatory body in the process of RWM Center establishment.

In year 2006 the radioactive waste characterization project in Krško NPP was performed. All LILW types of packages (28 of them) stored at Krško NPP were characterized by five basic properties (radiological, chemical, physical, mechanical and thermal). Additional four requirements (volume and mass, density of waste forms and packages, labelling and compliance with standards and practices in the field) were also taken into consideration. It should be noted that the main shortcomings of the characterization were the lack of data required by the preliminary WAC.

In order to evaluate the quantity of LILW stored in Krško NPP that can be packed in chosen concrete containers, the available data on LILW packages gathered by characterization, were compared with preliminary WAC. According to comparison results there are eight types of waste packages with different waste streams which do not meet preliminary WAC. Therefore, there is a need for their additional processing.

Keywords: *Radioactive Waste, Waste Acceptance Criteria, Waste Characterization, Waste Processing*

Session 8
Safety Culture, Knowledge Management and Public Relations (SCPR)

S8-121

33 years of the Nuclear Training Centre ICJT, Ljubljana

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Nuclear Training Centre Ljubljana ICJT has been operating for more than 30 years. Its main mission is training of future NPP operators and it is one of two institutions in Slovenia authorized to perform radiation protection training.

Practically all current control room operators in NPP Krško have begun their training at ICJT, as well as many others from technical support organizations, regulatory body, radwaste agency etc., both from Slovenia and Croatia.

In order to conduct all sorts of training in an effective and professional manner, a quality assurance system has been in place. This system is based on a set of procedures, databases of trainings, trainees, training materials and exam questions, as well as constant monitoring of feedback from trainees. Significant effort is also devoted to regular follow-up of modifications in NPP Krško and adequate updates of training materials.

An important advantage of training at ICJT is the use of TRIGA research reactor for practical exercises in nuclear physics, reactor physics and radiation protection. The use of TRIGA reactor also provides hands-on experience with a nuclear installation.

The paper will present the main achievements of ICJT in the last three decades, the QA system and the set of exercises on TRIGA. It will also provide an overview of the continuous training improvement process that is in place.

Keywords: *nuclear training, research reactor, quality assurance*

S8-125

Trust in scientists and experts on nuclear issues

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Scientific knowledge is an extremely important factor for general public to form an opinion on important issues, but also for policy makers in decision making process. Two factors affect scientific knowledge reaching the target: means of knowledge transfer and trust in scientific sources. Trust in science can be rated either based on trust in scientific methods and principles or based on trust in scientific institutions. Since believe in scientific principles and methods is generally high, the main focus of this manuscript is the latter one placed in the framework of nuclear issues. Particular attention is placed on utilization of nuclear energy for electricity generation and management of radioactive waste. Croatian case study is used to examine invoked topics. The analysis is based on the results of the national public opinion survey on nuclear energy and radioactive waste management carried out in 2016, and extensive observation of media coverage and public appearance of politicians and experts on the topics of interest. Preliminary findings indicate that the declarative level of trust in science in Croatia is generally high, in respect to the nuclear topics. However, the actions of targeted parties do not follow scientific recommendations. To further examine observed contradictions, the Croatian situation is placed in a wider European context.

Keywords: *trust in scientific knowledge, public opinion, policy makers*

S8-173

Young Generation Network of the Croatian Nuclear Society activities during period from 2020 to 2022

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The main aims of the Young Generation Network of the Croatian Nuclear Society (CNS YGN) are to inform and educate general public primarily about various applications of nuclear energy in peaceful purposes with emphasis on power system topics. Our members present facts from technical and scientific point of view while giving the audience overview of benefits and challenges of nuclear energy in order to make it more understandable and acceptable among general public.

The main activities of the CNS YGN consist of interactive lectures on nuclear energy that are given in Croatian high schools and during public events. Lectures usually consist of oral presentation followed by some demonstrations, such as the NPP Krško simulator, 3D models of nuclear reactor and fuel element, model of the control module, Geiger-Muller counter and an informative interactive application about radioactive waste. These exhibits gained positive feedback from people from pre-school children up to retired people.

In year 2020 our members took part in the Conference of Young Generation Network of the Slovenian Nuclear Society where they presented ongoing promotional activities. Also, they cooperated with staff from Technical Museum Nikola Tesla in Zagreb where they have filmed short videos that are part of museum's online content. Since 2020 was the 24th anniversary of nuclear accident in Chernobyl, our members provided media with quality information about the events that have led to this event.

In year 2021 most of the activities were conducted in an online format because of COVID-19 pandemics. Our members have given three online lectures and one in person for primary and secondary school students. Additionally, this was the 10th anniversary of nuclear accident in Fukushima and presentation about it have been held during Festival of Science. One of the major events organized by our members was the First Forum of Croatian Nuclear Society that took place from 30th of May to 2nd of June in city of Zadar. It has gathered more than 70 young experts mostly working in NPP Krško and large number of CNS members who have given lectures and participated in fruitful discussions.

In year 2022 members of CNS YGN have organized the Core Committee Meeting in Zagreb and continued with usual activities in schools. Furthermore, our members closely cooperated with members of Fund for Financing the Decommissioning of the Krško NPP and helped to create promotive materials on the topic of radioactive waste that will be exhibited in Technical Museum Nikola Tesla in Zagreb. These

are just some of the activities conducted by members of CNS YGN and many more are in preparation for future events.

***Keywords**: Young Generation Network, Croatian Nuclear Society, nuclear promotion, communicating nuclear, nuclear exhibition*

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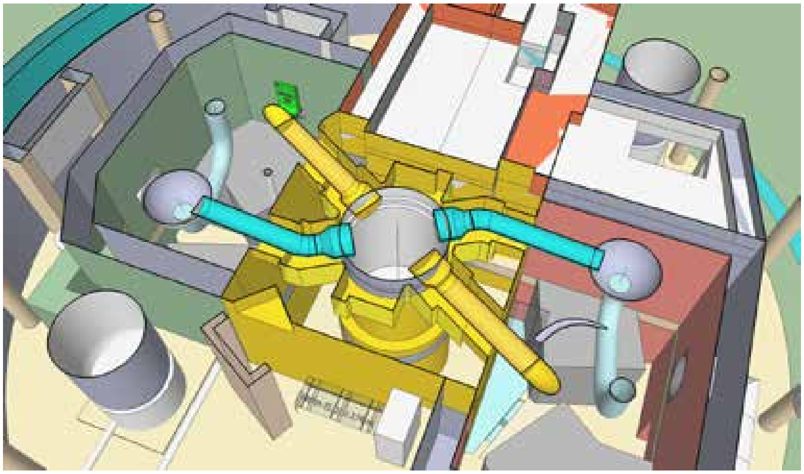

Croatian Nuclear Society

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