NENE 2024
September 9 - 12
PORTOROŽ
SLOVENIA
33rd International Conference Nuclear Energy for New Europe

BOOK OF ABSTRACTS

Nuclear Research: Securing a Predictable Future

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Nuclear Society of Slovenia

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Ljubljana, Slovenia

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NENE

BOOK OF ABSTRACTS

of 33rd International Conference Nuclear Energy for New Europe
titled Nuclear Research: Securing a Predictable Future

September 9-12, 2024
Portorož, Slovenia

https://www.djs.si/nene2024

Publisher	Nuclear Society of Slovenia, Jamova cesta 39, 1000 Ljubljana, Slovenia djs@djs.si www.djs.si
Editors	Luka Snoj Tanja Goričanec
Cover Photo	Bojan Žefran
Design	Lenka Trdina Marjeta Trobec
Edition	180 copies
Printed by	Birografika BORI d.o.o.

CIP - Kataložni zapis o publikaciji
Narodna in univerzitetna knjižnica, Ljubljana

621.039(4)(082)

INTERNATIONAL Conference Nuclear Energy for New Europe (33 ; 2024 ; Portorož)
Book of abstracts of 33rd International Conference Nuclear Energy for New Europe titled Nuclear Research: Securing a Predictable Future : NENE : September 9-12, 2024, Portorož, Slovenia / [editors Luka Snoj, Tanja Goričanec]. - Ljubljana : Nuclear Society of Slovenia, 2024

ISBN 978-961-6207-58-4
COBISS.SI-ID 203782147

This project was funded, in part, through a U.S. Embassy grant. The opinion, findings, and conclusions or recommendations expressed herein are those of the Authors and do not necessarily reflect those of the Department of State.

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Previous meetings organized by the Nuclear Society of Slovenia

- First Meeting of Nuclear Society of Slovenia, Bovec, Slovenia, June 1992
- Regional Meeting: Nuclear Energy in Central Europe, Present and Perspectives,
- Portorož, Slovenia, June 1993
- PSA/PRA and Severe Accidents ‘94, Ljubljana, Slovenia, April 1994
- Annual Meeting of NSS ‘94, Rogaška Slatina, Slovenia, September 1994
- 2nd Regional Meeting: Nuclear Energy in Central Europe, Portorož, Slovenia, September 1995
- 3rd Regional Meeting: Nuclear Energy in Central Europe, Portorož, Slovenia, September 1996
- 4th Regional Meeting: Nuclear Energy in Central Europe, Bled, Slovenia, September 1997
- Nuclear Energy in Central Europe `98, Čatež, Slovenia, September 1998
- Nuclear Energy in Central Europe `99 with Embedded Meeting Neutron Imaging Methods to Detect Defects in Materials, Portorož, Slovenia, September 1999
- 20th International Conference on Nuclear Tracks in Solids, Portorož, Slovenia, August 2000
- Nuclear Energy in Central Europe 2000, Bled, Slovenia, September 2000
- Nuclear Energy in Central Europe 2001, Portorož, Slovenia, September 2001
- Nuclear Energy for New Europe 2002, Kranjska Gora, Slovenia, September 2002
- Nuclear Energy for New Europe 2003, Portorož, Slovenia, September 2003
- Nuclear Energy for New Europe 2004, Portorož, Slovenia, September 2004
- Nuclear Energy for New Europe 2005, Bled, Slovenia, September 2005
- Nuclear Energy for New Europe 2006, Portorož, Slovenia, September 2006
- Nuclear Energy for New Europe 2007, Portorož, Slovenia, September 2007
- Nuclear Energy for New Europe 2008, Portorož, Slovenia, September 2008
- Nuclear Energy for New Europe 2009, Bled, Slovenia, September 2009
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- Nuclear Energy for New Europe 2017, Bled, Slovenia, September 2017
- Nuclear Energy for New Europe 2018, Portorož, Slovenia, September 2018
- Nuclear Energy for New Europe 2019, Portorož, Slovenia, September 2019
- Nuclear Energy for New Europe 2020, Portorož, Slovenia, September 2020
- Nuclear Energy for New Europe 2021, Bled, Slovenia, September 2021
- Nuclear Energy for New Europe 2022, Portorož, Slovenia, September 2022
- Nuclear Energy for New Europe 2023, Portorož, Slovenia, September 2023

Welcome

Dear participants,

Welcome to the 33rd International Conference Nuclear Energy for a New Europe, an important gathering of experts addressing the future of nuclear energy in our rapidly evolving world. As we gather in this forum, we are reminded of the profound impact nuclear energy has on our society, economy and environment.

The conference is an annual gathering of nuclear research and education professionals, nuclear vendors, utilities and regulators with a 33-year tradition. It attracts participants from all over Europe and the world. The broad range of topics includes advances in nuclear technology, reactor physics and research reactors, severe accidents, nuclear power plant operation, thermohydraulics and CFD, nuclear regulations, nuclear fusion and plasma technology, materials and ageing management, the environment and the back end of the fuel cycle, and education, training and outreach.

This conference is not only a platform for knowledge exchange, but also a crucible for new collaborations and partnerships that will advance our shared mission. This year's conference theme is "Nuclear Research: Securing a Predictable Future". During the approximately 100-year lifespan of a nuclear power plant, humanity undergoes numerous social, demographic and technological changes. The only way to meet these changes is through continuous research and development. Research is the only human activity capable of predicting the future.

New breakthroughs, challenges and opportunities in the nuclear sector will be highlighted during the four-day conference. Six invited speakers will share their insights on exciting nuclear topics. In addition, 41 oral presentations in 13 plenary sessions covering a wide range of topics will be complemented by 92 posters presented during coffee breaks. In parallel, a special side event SMR Stakeholder Event will focus on small modular reactors. The event will be coordinated by the Slovenian members of the European Industrial Alliance on SMRs.

The traditional competition for young authors will be a great opportunity for the new generation of experts to present their ideas and work. The best posters will also be awarded.

In tackling the complex challenges of the 21st century, the role of nuclear energy in achieving a sustainable and low-carbon future cannot be overemphasised. The discussions and insights gained at this conference will undoubtedly help to develop policies and practises that ensure the safe, efficient and equitable use of nuclear energy.

Today, it is time to understand more so that we have less to fear. Therefore, investing in education and science and disseminating scientific knowledge and facts to all parts of society is extremely important. NENE 2024 has an important role to play in this endeavour.

Thank you for joining us in this important endeavour. We look forward to the fruitful exchange and collaboration that will emerge from this conference.

Dr. Tanja Goričanec
Organizing committee chair

Prof. Dr. Luka Snoj
Program committee chair

Place and time of conference

The conference will take place in Grand Hotel Bernardin in Portorož, Slovenia:

St. Bernardin Resort
Grand Hotel Bernardin
Obala 2
6320 Portorož

From: Monday, September 9, at 15:00
To: Thursday, September 12, at 16:20

Lectures and poster sessions will take place on the 12th floor of Grand Hotel Bernardin.

Registration desk

Information and conference materials are available at the registration desk.

Opening hours:

Monday, September 9:	from 9:00 to 10:00 and from 13:00 to 18:00
Tuesday, September 10:	from 8:00 to 18:00
Wednesday, September 11:	from 8:00 to 12:00
Thursday, September 12:	from 8:00 to 12:00

Presentations

Oral presentations are limited to 20 minutes. Authors are asked to present their papers in **15 minutes** and allow 5 minutes for discussion. A laptop with Microsoft PowerPoint will be available.

Short oral presentations are limited to 5 minutes. Young authors are asked to present their contributions in **3 minutes** and allow 2 minutes for discussion.

Posters should fit within the **95 cm (width) × 110 cm (height)** frame. Authors are asked to put up their posters by Tuesday at 15:20 and remove them by Thursday at 14:00 at the latest.

The best poster presentation will be awarded a special prize.

Publications

Book of Abstracts
Each participant will receive the Book of abstracts in printed form. The Book of abstracts will also be available in electronic form on the conference website.

Proceedings
Proceedings containing full length peer review papers presented at the conference will be published in electronic form after the conference and sent to participants. The proceedings will also be available in electronic form on the conference website.

Awards

Young Author Award Contest
A prize will be awarded for the best paper prepared by a first author who is not older than 32 years. Young Authors will present a short oral presentation at the conference and a poster in the corresponding poster section. The Young Authors Award Committee will review the eligible papers and select the winner.

Best Poster Award
The best poster will be selected by the Best Poster Award Committee. Posters will be evaluated according to clarity of objectives, results and conclusions, scientific relevance, aesthetics and attractiveness.

Social activities

Conference Lunch on Tuesday, Wednesday and Thursday
Lunch is included in the registration fee and will be served from 12:30 to 14:00 in the Grand Restaurant on the 10th floor of the Grand Hotel Bernardin.

Welcome Reception, Monday, September 9
The welcome reception starts at 19:30 on the Terrace International, which is located on the square with the church between the Vile Park and Histrion hotels. In case of bad weather, the welcome reception will take place on the Grand Garden Terrace on the 11th floor of the Grand Hotel Bernardin.

Conference Trip, Wednesday, September 11
The conference trip will take place on Wednesday, September 11, at 14:30.
We will meet in the hotel lobby and set off. The boat will pick us up at the chosen pier and take us to the panorama of Slovenian coastal towns and other small bays along the Riviera.

Conference Dinner, Wednesday, September 11
The conference dinner will take place on Wednesday at 19:30 at Sunset Restaurant on the 10th floor of the Grand Hotel Bernardin. During the dinner the Young Authors Award and the Best Paper Award will be presented.

Special Events

SMR Stakeholder Event, Monday, September 9
An SMR Stakeholder Engagement event will take place on Monday from 10:00 to 13:30 on the 12th floor of the Grand Hotel Bernardin. The SMR Stakeholder Engagement event is organised as a hybrid activity in the run-up to NENE 2024 and welcomes both local and remote participants. The event is coordinated by the Slovenian members of the European Industrial Alliance on SMRs.

Group photo, Tuesday, September 10
The group photo will be taken on Tuesday, September 10, at 12:30 in front of the entrance of the Grand Hotel Bernardin.

Young Author Contest Award Ceremony
The award ceremony will take place on Wednesday, September 11, during the conference dinner. The chair will present the decision of the Young Author Contest Committee and award the prize to a maximum of two authors.

Best poster Award Ceremony
The award ceremony will take place on Wednesday, September 11, during the conference dinner. The chair will present the Best Poster Award Committee decision and give the award to maximum of two authors.

Technical tour to JSI TRIGA, Friday, September 13
On Friday morning, a technical tour of the Jožef Stefan Institute, TRIGA Reactor, Ljubljana will be organised. Transportation will pick you up at 8:00 am in front of Hotel Bernardin. The tour of the TRIGA reactor of the JSI is scheduled between 10:00 and 12:00. Lunch will be provided for tour participants between 12:00 and 13:00. Return to the hotel at 15:00.
The technical tour includes transportation to and from the hotel, lunch and a guided tour of the JSI TRIGA reactor.

Conference Timetable

The oral presentations will take place in **Europa A** and the poster sessions in **Europa C** on the 12th floor of the Grand Hotel Bernardin

	Monday Sep. 9	Tuesday Sep. 10	Wednesday Sep. 11	Thursday Sep. 12	Friday Sep. 13
8:00		*Registration*	*Registration*	*Registration*	TRIGA technical tour
8:30		Invited	Invited	Invited	
9:10		Research reactors	Thermal-hydraulics, computational fluid dynamics	Nuclear fusion and plasma technologies	
10:10	SMR Stakeholder Engagement Event (10:00 - 13:30)	*Coffee Break* Reactor physics	Posters with coffee break Materials in nuclear technology	Posters with coffee break Nuclear power plant operation & plant life management	
12:30		*Lunch*	*Lunch*	*Lunch*	
14:00	*Registration*	Fuel cycle, radioactive waste, and decommissioning	Conference trip	Radiation and env. protection	
15:00	*Opening ceremony* Invited NPP suppliers *Coffee Break* New reactor designs and small modular reactors	Poster with coffee break Safety analyses, Severe accidents and PSA		Education and training Regulatory issues and legislation *Closing ceremony*	
19:30	*Welcome reception*		*Conference dinner*		

Preliminary Program of the
International Conference NENE2024

Monday, Sept. 9

15:40 Invited lectures

No. 1 15:40
EDF's approach to deliver Slovenia's nuclear ambition
Anne Falchi - EDF, France

No. 2 16:00
KHNP APR1000: Optimal Design Solution adaptable for JEK2 project
Keunho Lee - KHNP, Korea

No. 3 16:20
Harnessing the Power of the Nth of a Kind AP1000 Technology Global Deployment to Strengthen Slovenia's Nuclear Future
Thomas Weir - Westinghouse Electric Company, United States of America

17:00 New reactor designs and small modular reactors

No. 501 17:00
Fuel for the Future – Two Reactor Concepts to Tackle the Burden of Accumulating Spent Fuel and High-level Radioactive Waste
Mehmet Kadiroglu, Mohammad Hessan, Kai-Martin Haendel - Germany

No. 502 17:20
Heat-Only Small Modular Reactors Vs. Nuclear Combined Heat and Electricity: Concepts and Economics
Hussein Abdulkareem saleh Abushamah, Ondrej Burian, Radek Skoda - Czech Republic

No. 503 17:40
Development of small modular reactor cooled by supercritical water within the ECC-SMART project
Monika Šípová, Alberto Sáez-Maderuelo, Ivan Otic, Szabolcs Czifrus, Leon Cizelj - Czech Republic

No. 504 18:00
3D Reactor Core Simulation of ACP100 SMR
Ye Zhu, Wang Xingbo - China, People's Republic of

No. 505 18:20
Review of Existing and Emerging Nuclear New Build Business Models
Ana Stanič - United Kingdom

No. 506 18:40
Preliminary Design and Optimization of the Heat Pipe Heat Exchanger (HPHX) for a 5 MWth Heat Pipe-Cooled Micro Modular Reactor (MMR)
Matthias Peiretti, Michael Buck, Jörg Starflinger - Germany

Tuesday, Sept. 10

8:30 Invited lecture

No. 4 8:30
Lecture No. 4

9:10 Research reactors

No. 201 9:10
The use of low power research reactors to develop new protocols in the pursuit of exotic radionuclide production
Giorgio Grosso, Andrea Salvini, Andrea Gandini, Daniele Alloni, Federico Alfinito, Letizia Canziani – Italy

No. 202. 9:30
Review of SiC based sensor performances for neutron flux measurements in nuclear reactor fission facilities
Christophe Destouches, Quentin Potiron, Olivier Llido - France

No. 203 9:50
Shadowing Effect Correction for the Pavia TRIGA Reactor Using Monte Carlo Data and Reduced Modelling Techniques
Cyrille Ghislain De Lurion De L'Égouthail, Lorenzo Loi, Stefano Riva, Carolina Introini, Antonio Cammi - Italy

No. 204 9:55
Enhancing Irradiation Facilities: A Study on the Upgraded Water Filtration System in the KATANA Water Activation Loop
Julijan Peric, Domen Kotnik, Domen Govekar, Luka Snoj, Vladimir Radulović - Slovenia

10:30 Reactor physics

No. 101 10:30
Initial development of a multi-point kinetic model for a small modular heat-only reactor TEPLATOR
Dipanjan Ray, Martin Lovecký, Jiří Závorka, Radek Škoda - Czech Republic

No. 514 10:50
Nuclear jumps into space exploration challenges
Grégoire Lambert - Framatome, France

No. 103 11:10
Analysis of KRUSTY reactor behaviour with OFELIA environment
Riccardo Boccelli, Lorenzo Loi, Stefano Riva, Carolina Introini, Stefano Lorenzi, Antonio Cammi - Italy

No. 104 11:30
Current improvements to evaluated nuclear data from the INDEN project
Andrej Trkov, Roberto Capote - Slovenia

No. 105 11:50
On the Impact of Cross Section and Fission Product Yield Data on PWR Design Calculations with CORD-2
Jan Malec, Andrej Trkov, Marjan Kromar - Slovenia

No. 106 11:55
Parameterized cross-section library generation with the SERPENT code using ML methods and testing in VVER-1200 core geometry
Dániel Sebestény, István Pataki, István Panka - Hungary

No. 107 12:00
Application of Machine Learning techniques to the exploitation and interpolation of nuclear experimental data
Adèle Berger, Ivan Kodeli, Pierre-Jacques Dossantos-Uzarralde - France

No. 108 12:05
Sensitivity analysis of the Krško NPP internal and external components to the activation of material impurities
Benjamin Barbarič, Tanja Goričanec, Klemen Ambrožič, Luka Snoj - Slovenia

No. 110 12:10
Adaptive Design and Criticality Analysis of the DARWIN Reactor Core Concept
Anže Mihelčič, Dušan Čalič, Luka Snoj - Slovenia

No. 111 12:15
Investigating the potential fuels for DFRm reactor concept
Semra Daydaş, Ali Tiftikçi - Turkiye

14:00 Fuel cycle, radioactive waste, and decommissioning

No. 601 14:00
Status of Slovenian LILW disposal facility
Špela Mechora, Mateja Zupan, Sandi Viršek - Slovenia

No. 602 14:20
Radioactive Waste Disposal in Germany
Eileen Langegger, Marie Charlotte Bornhöft - Germany

No. 603 14:40
Exposing Iron-rich Inorganic Polymers to Cobalt-60 radiation: Leaching behavior of Cs, Sr, and Eu nitrates and structural changes of the material
Evangelia Dimitra Mooren, Stefaan Van Winckel, Rafael Alvarez- Sarandes, Walter Bonani, Andrea Cambriani, Glenn Beersaerts, Vaclav Tyrpekl, Tomas Cernousek, Sonja Schreurs, Rudy Konings, Wouter Schroeyers - Germany

No. 604 15:00
Validation of the improved Siempelkamp NIS 3D activation calculation method for nuclear decommissioning
Imrich Fabry, Gerhard Graebner, Robert Holzer - Germany

No. 605 15:20
Automated object detection and position extraction of legacy nuclear waste using a robotic manipulator
Alex Jackie Carpenter, David Megson-Smith, Thomas Bligh Scott - United Kingdom

16:20 Safety analyses, Severe accidents and PSA

No. 401 16:20
The CHIP Line: an unique experimental device dedicated to fission product chemistry and transport at high temperature in PWR severe accident conditions
Anne Cécile Gregoire, William Le Saux, Calogero Tornabene, Guillaume Bourbon, Sandrine Morin, Laurent Cantrel - France

No. 402 16:40
Decay heat removal from the MSFR core through the passive safety system
Barbara Kędzierska, Alexis Saint-Dizier, Xue-Nong Chen, Andrei Rineiski - Germany

No. 403 117:00
Assessment of ATHLET code through its application to a LFR decay heat removal system
David Giron Ceballos, Donella Pellini, Barbara Calgaro - Italy

No. 404 17:20
Assessment of TRACE V5.0 Patch 8 using BETHSY 4.1a TC test
Andrej Prošek - Slovenia

No. 405 17:40
Probabilistic margin assessment for PTS analysis during LTO
Yaroslav Dubyk - Ukraine

No. 406 18:00
Non-ergodic ground motion model for NPP sites in Slovenia
Anže Babič, Norman Abrahamson, Matjaž Dolšek - Slovenia

No. 407 18:20
Improved HPC models for fuel performance predictions
Salvatore Angelo Cancemi, Rosa Lo Frano, Michela Angelucci, Riccardo Ciolini - Italy

No. 408 18:40
Application of AI methods for describing the coolability of debris beds formed in the late accident phase of nuclear reactors
Jasmin Joshi-Thompson, Michael Buck, Joerg Starflinger - Germany

No. 409 19:00
Evolution of Safety Concepts in Pressurized Water Reactors
Daniel Hackl, Eileen Langegger, Helmuth Böck - Austria

Wednesday, Sept. 11

8:30 Invited lecture

No. 5 8:30
Interface Capturing Simulations of Nuclear Reactor Flows
Igor Bolotnov - North Carolina State University, United States of America

9:10 Thermal-hydraulics, computational fluid dynamics

No. 301 9:10
Potential effect of saline solution on fission products mitigation by pool scrubbing
Fouzia Djeriouat, Catherine Marchetto, Philippe Nerisson, Maxime Chinaud, Olivier Vauquelin - France

No. 302 9:30,
Application of system and CFD codes on simulation of PERSEO experiment passive heat removal systém within the EU PASTELS project
Adam Kecek, Ladislav Vyskocil - Czech Republic

No. 303 9:50
Exploring Taylor Bubble Dynamics in Counter-Current Flows: A Combined Numerical and Experimental Study
Jan Kren, Iztok Tiselj, Blaž Mikuž - Slovenia

No. 304 9:55
Design and construction of the transparent test section for analysis of boiling flow scaled to fusion divertor conditions
Jakob Jakše, Gregor Kozmus, Boštjan Končar - Slovenia

No. 305 10:00
Lagrangian simulation of flow boiling experiments in horizontal annulus
Boštjan Zajec, Žiga Perne, Boštjan Končar - Slovenia

No. 307 10:05
Influence of Chemical and Physical Treatments of Copper Surface on the Contact Angle of R-245fa Refrigerant
Deja Razpet, Žiga Tominc, Blaž MIkuž - Slovenia

11:10 Materials in nuclear technology

No. 701 11:10
Development of irradiation and high-temperature resistant steels for nuclear applications
Dmitry Terentyev - Belgium

No. 702 11:30
Stress Corrosion Cracking of AM 316L in PWR Primary Water
Radek Novotny, Michal Novak, Jan Siegl, Monika Sipova, Oliver Martin - Czech Republic

No. 703 11:50

Chemical production via radiolysis: chemical reactor design optimization

Klemen Ambrožič, Vladimir Radulović, Luka Snoj – Slovenia

No. 704 12:10

The FFT-Homogenization Method in Irradiated Austenitic Stainless Steels

Amirhossein Lame Jouybari, Samir El Shawish, Leon Cizelj - Slovenia

No. 705 12:15

Crack growth prediction with phase-field method

Patrik Tarfila, Oriol Costa Garrido, Mitja Uršič - Slovenia

No. 706 12:20

Sounding Out Separation: Numerical Investigation of Utilizing Ultrasound for Medical, Fission, and Fusion Isotope Enrichment

Ena Karić, Klemen Ambrožič - Bosnia and Herzegovina

Thursday, Sept. 12

8:30 Invited lecture

No. 6 8:30
Current status and challenges of the development of plasma facing components for fusion devices
Rudolf Neu - Germany

9:10 Nuclear fusion and plasma technologies

No. 1001 9:10
Construction and commissioning of the KATANA water activation loop at the JSI TRIGA research reactor
Domen Kotnik, Julijan Peric, Domen Govekar, Luka Snoj, Igor Lengar - Slovenia

No. 1002 9:30
Detection of defects and deuterium in displacement-damaged tungsten by ion beam methods in channeling configuration for fusion application
Sabina Markelj, Esther Punzón-Quijorna, Mitja Kelemen, Thomas Schwarz-Selinger, Xin Jin, Eryang Lu, Flyura Djurabekova, Kai Nordlund, Janez Zavašnik, Andreja Šestan, Miguel L. Crespillo, Gaston García López, Rene Heller - Slovenia

No. 1003 9:50
Comparative Analysis of SDDR Calculation Approaches in simplified Large-Scale Tokamak Models
Ylenia Kogovšek Žiber, Igor Lengar, Klemen Ambrožič - Slovenia

No. 1004 9:55
Analysis of four heat flux partitioning model at divertor cooling conditions
Aljoša Gajšek, Matej Tekavčič, Boštjan Končar - Slovenia

No. 1005 10:00
Thermo-Mechanical Study of Different Solutions for the Inner Top FW Module of DTT
Nejc Kromar, Oriol Costa Garrido, Maurizio Furno Palumbo, Selanna Roccella - Slovenia

No. 1006 10:05
Visualization experiment for analysis of two-phase flow in divertor cooling channels
Gregor Kozmus, Jakob Jakše, Boštjan Končar - Slovenia

10:50 Nuclear power plant operation & plant life management

No. 1101 10:50
Assessment of Selected Long-Term Operation Improvements Relevant to the Pressurized Thermal Shock in PWR with Focus on Reactor Pressure Vessel Integrity
Piotr Darnowski, Piotr Emil Mazgaj, Miroslav Pošta - Poland

No. 1102 11:10

Assessment of the optimal electric power of the new nuclear power plant JEK2 for secure and stable operation and development of the electric power system of Slovenia

Aleksandar Momirovski, Luka Zidarič, Maja Kernjak Jager, Ana Gjorgjovska, Miloš Maksić, Igor Podbelšek, Jurij Kurnik, Robert Bergant, Bruno Glaser, Dejan Paravan, Klemen Dragaš, Tomaž Tomšič, Aljoša Deželak, Nikola Rebić, Marko Kolenc - Slovenia

No. 1103 11:30

Cable Aging Management Program and Development of Acceptance Criteria for Radiation Aged Field Assessment of Polymers in Krsko Nuclear Power Plant (NEK)

Marko Pirc, Luka Snoj - Slovenia

No. 1104 11:50

Indicative pre-investment economic analysis of the project JEK2

Jan Lokar, Robert Bergant, Kruno Abramovič, Tomaž Žagar - Slovenia

No. 1105 12:10

SI-53 direct cause analysis report

Stanko Manojlovič - Slovenia

14:00 Radiation and environmental protection

No. 801 14:00

Evaluation of Average Natural Background Radiation Dose in Slovenia

Andrej Žohar, Marko Giacomelli, Peter Jovanovič, Gregor Omahen, Manca Podvratnik, Matija Škrlep - Slovenia

No. 802 14:20

The Utilisation of Legacy Mine Sites as Training Environments for Radiological Emergency Response Preparedness

Ewan Woodbridge, Billy Murphy, Dean T Connor, Yannick Verbelen, David Megson-Smith, Thomas Bennett, Sofia Leadbetter, Thomas B Scott - United Kingdom

No. 803 14:25

Preliminary Results of the Dudváh River Contamination from the 1977 A1 Nuclear Reactor Accident

Otto Glavo, Branislav Vrban, Jakub Lüley, Vendula Filová, Vladimír Nečas - Slovak Republic

No 804 14:30

Calibration of HPGe detectors and usage of prompt gamma rays to extend detector efficiency curve in >2 MeV energy range

Domen Govekar, Julijan Peric, Domen Kotnik, Vladimir Radulović - Slovenia

Education and training and public outreach

No. 901 14:40

International Higher Education with I2EN and the French Nuclear Industry

Jan van der Lee - France

No. 902 15:00

Striving for Excellence: Improving Plant Systems Refresher Training at NPP Krško

Matjaž Žvar – Slovenia

Regulatory issues and legislation

No. 1201 15:20

Licensing procedure for a new nuclear power plant JEK2

Sonja Torkar, Barbara Vokal Nemec, Tomaž Nemec, Špela Krajnc, Benja Režonja Gumpot, Tomi Živko – Slovenia

No 1202 15:40

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EDF's approach to deliver Slovenia's nuclear ambition

Anne Falchi
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In the face of climate change, soaring energy prices and energy security concerns, more and more countries are turning to nuclear. Nuclear not only provides clean and sustainable energy and its safe supply of electricity; it also guarantees security of supply and represents a real potential for national energy independence. Many European countries, as Slovenia, are putting in place nuclear programs, including extending the life duration of existing ones, ensuring the resilience of the fuel supply chain to changing geopolitical conditions, or launching the construction of new build programs. Fully committed to the Slovenian nuclear programme, EDF together with the French and European nuclear industry, is providing a European solution to these challenges, based on four pilars:

- a European, robust and proven technology enabling the owner to meet its strategic independence;
- a European supply chain, with 95% of the value of our on-going projects in Europe;
- a European solution for the fuel supply;
- a collaborative approach & a fleet effect to deliver the best projects for Europe, by developing a portfolio of products designed for the European needs.

2
Invited lectures

KHNP APR1000: Optimal Design Solution Adaptable for JEK2 project

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The APR1000, a Generation III+ Pressurized Water Reactor (PWR) with a 1,000 MWe capacity, meets the top safety and performance standards. It integrates proven technologies from APR1400 and OPR1000 platforms, drawing from international safety protocols like IAEA, WENRA, and EUR. By incorporating modern design features from the APR1400, it enhances safety and performance, while its core design and RCS configuration originate from the OPR1000. The primary design philosophy includes safety enhancement, technology utilization, and adaptable standardized design. KHNP's proposed APR1000 references Saeul Units 3 and 4, exemplary Generation III pressurized water reactors currently undergoing commissioning in Korea. With a rich legacy spanning 50 years of continuous construction and operation of nuclear power plants, Korea operates twenty-six units and is constructing two APR1400 units domestically. Additionally, three APR1400 units are operational in the UAE, with another undergoing commissioning. This plant obtained certification from the EUR organization and design certification from the US NRC in November 2017 and August 2019, respectively. In March 2023, the APR1000 also obtained EUR certification based on the latest EUR standards. Leveraging extensive experience, the APR1000 is poised to be the optimal solution for the second Krško Nuclear Power Plant (JEK2), seamlessly integrating proven capabilities.

3
Invited lecture

Harnessing the Power of the Nth of a Kind AP1000 Technology Global Deployment to Strengthen Slovenia's Nuclear Future

Thomas J. Weir
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The presentation will highlight Westinghouse's product portfolio with a focus on the AP1000 technology, a cornerstone of nuclear energy innovation. We will showcase the project successes of the first 6 AP1000 units in operation, highlighting their safety, efficiency, and reliability. Drawing from this first wave of AP1000 deployments, we will also share insights surrounding delivery certainty of nuclear new builds, specific to the upcoming Slovenia project. Additionally, we will discuss the broad range of local and global benefits of expanding baseload nuclear capacity in Slovenia

5
Invited lecture

Interface Capturing Simulations of Nuclear Reactor Flows

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Interface Capturing simulations are becoming a more practical tool for complex flow analysis due to significant improvement of flow solvers, pre- and post-processing tools as well as rapid development of high-performance computing capabilities. This creates exciting opportunities to study complex reactor thermal hydraulic phenomena. This presentation will focus on the history and review of numerical flow simulation approaches in recent years, capabilities development and validation as well as the applications to practical problems of interest. We will discuss typical computational methods used for those simulations, provide some examples of past work, as well as computational cost estimates and affordability of such simulations for research and industrial applications. New generation methodologies are required to take full advantage of those capabilities to greatly enhance the scientific understanding of complex flow phenomena in various conditions relevant to nuclear energy applications.

6
Invited lecture

Current status and challenges of the development of plasma facing components for fusion devices

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In view of the severe operating conditions for plasma facing components (PFCs) in future power producing fusion devices, the development of advanced components and materials is mandatory. The PFCs not only have to withstand high steady state power loads but also a high number of thermal cycles and shocks. Moreover, the change of thermo-mechanical properties by lattice damage, activation and transmutation through fusion neutrons has to be considered when designing PFCs and selecting adequate armour and structural materials. Presently, water-cooled PFCs are foreseen in all future fusion devices in order to provide reliable heat removal capability and to only moderately extrapolate the technologies developed and tested for ITER. However,

attempts were undertaken to optimize the design as well as the armour and heat sink materials in view of future applications with even harsher conditions. The contribution will give an overview on the requirements for plasma facing components and the state-of-the-art solutions. In addition, new concepts and materials will be presented which should be capable of facing the challenges in future fusion reactors.

Reactor physics

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Reactor physics

Initial development of a multi-point kinetic model for a small modular heat-only reactor TEPLATOR

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Nuclear reactor design and safety analysis rely on the utilization of detailed system codes which can capture the dynamic behaviour of the system in accurate way. These codes employ time dependent coupling between detailed neutronic and multi-component thermal hydraulic model for that purpose. However, inherent complexity of the system codes makes the calculation quite tedious in case of identifying the overall visualization of the parameter space in reactor design related analysis. Therefore, it is of interest to develop a code that can maintain balance between simplicity and adequacy, enabling the simulation of design basis and design extension classified transients in new conceptual reactors. The point kinetics approach falls short in analysing the asymmetrical behaviour of large-dimension reactor cores due to neutronic loose coupling. This oversimplification of the actual dynamics of the reactor core can be compensated by using multi-point kinetic approach. In this approach, reactor core is divided into multiple regions or nodes, each with its own set of neutronic parameters. This approach allows for a more detailed and realistic simulation of the core behaviour, capturing spatial variations and transient effects that can significantly impact the performance and safety of the reactor. Additionally, the multi-point kinetic model is well-suited for conceptual design studies. During the early stages of reactor design, when multiple design options and configurations are being explored, the multi-point kinetic model provides a simplified representation of the reactor core, allowing for rapid evaluation of different core configurations, fuel compositions, and operating conditions. This approach expedites decision-making and offers valuable insights before committing extensive resources to detailed neutronic calculations. In this article, initial stage of developing the multi-point kinetic model of a heavy water small modular reactor concept TEPLATOR is discussed. For the current analysis, only the core regions are considered. Mathematical model of the reactor core within the framework of nodal modeling is derived with the multi-group neutron diffusion equation as a basis. This multi-group neutron diffusion model is developed using COMSOL Multi-physics software package, while the group constants for the diffusion equation are calculated using Monte Carlo transport code Serpent version 2.2.1 with the latest ENDF/B-VIII.0 nuclear data library. During the first stage of analysis, simulation is conducted considering the overall design on the reactor core, including the position and dimensions of fuel assemblies and control rods. These simulations aim to determine an optimal nodalization scheme that combines axial and radial divisions, ensuring the prediction of overall core behaviour with sufficient accuracy. This analysis will serve the foundation for developing a more detailed model that incorporates heat transfer between core components and, also the option to include the primary circuits of the plant. Detailed multi-point model will help to evaluate stability and robustness of the dynamic model and, as well as examining the impact of the fundamental parameters such as feedback coefficient of reactivity to the system stability, both for a stand-alone core and in a primary loop configuration.

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Reactor physics

Analysis of KRUSTY reactor behaviour with OFELIA environment

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Mathematical modeling plays a crucial role in designing and obtaining licenses for nuclear reactor cores, particularly when many physics are involved e.g. neutronics, thermo-mechanics, thermal hydraulics etc. This becomes even more significant for innovative reactors like KRUSTY, which uses unique heat-pipe cooling and Uranium-Molybdenum monolith blocks as fuel. KRUSTY serves as a testbed for future space reactors, where environmental conditions are harsh and human intervention is limited.

The study analyzes the stationary behavior of the KRUSTY reactor under different power levels aiming at capturing the coupling between the neutronics and the thermal phenomena within the OFELIA environment. OFELIA is a Python-based open-source multiphysics environment based on OpenMC and FEniCSx codes. Despite its design simplicity, the analysis of the KRUSTY reactor requires an accurate evaluation of the temperature and power distribution inside the fuel structure in order to assess the reactivity feedback coefficient with high accuracy, allowing for the evaluation of passive safety and the autonomous load following capabilities of the reactor. In this regard, the coupling framework provided by OFELIA is essential. The results of the coupled model are evaluated and compared against the experimental results obtained during the KILOPOWER project in order to validate the model and to assess the accuracy of the proposed simulation approach.

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Reactor physics

Current improvements to evaluated nuclear data from the INDEN project

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The CIELO project of the OECD/NEA Data Bank was a major effort to improve the evaluations of the major nuclides for fission reactor technology. Many of the evaluations were adopted for the ENDF/B-VIII.0 library, which was released in 2018. The library shows significant improvement compared to its predecessor ENDF/B-VII.1. Since then, additional improvements to specific evaluations were made, which have a relatively smaller impact on the suite of benchmarks from the ICSBEP collection, like the Mosteller suite, but show significant progress in cases with a high sensitivity to iron, chromium, copper, fluorine, silicon and Pu-239. Improvement were also made in describing the reactivity loss with burnup, which showed a somewhat degraded performance with ENDF/B-VIII.0, compared to the previous ENDF/B-VII.1 library. The improved evaluations are candidates for inclusion in the ENDF/B-VIII.1 and JEFF-4.0 libraries, which are scheduled for release later this year.

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Topics: Reactor physics

On the Impact of Cross Section and Fission Product Yield Data on PWR Design Calculations with CORD-2

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In response to user feedback regarding the burnup gradient in depletion calculations in the latest incident neutron nuclear data libraries, we initiated an investigation into reactivity loss reproduction using different computational codes. We found that reactivity loss can be consistently reproduced across all codes, but this consistency depends not only on cross sections but also on the energy released per fission and other factors. We employed two models in our study: the UAM_TMI-1 (a Three Mile Island pin cell model) and cycle-dependent boron concentrations at different core burnup steps of the Krško NPP, calculated with CORD-2 and compared to measured data.

For the UAM-TMI-1 pin-cell benchmark, our findings indicate that different codes, including Serpent, OpenMC, WIMS-D, Dragon, and VESTA, can predict reactivity loss at 50 GWd/tU within 100 pcm of each other, when similar assumptions about energy per fission (EDEP) are made. Discrepancy with WIMS-D results is about 200 pcm mainly due to the inability to change the EDEP model. Serpent and OpenMC both provide more than one energy deposition model for neutron transport calculations and different ways of accounting for energy from capture-gamma rays. Serpent normalizes fission Q-values to account for capture gammas, while by default, OpenMC directly uses the fission Q values from the ENDF file, which underestimates the energy release. Forcing the fission Q values in OpenMC to match the values used in Serpent restores the agreement. Furthermore, increasing the value by 1% improves agreement with WIMS-D, indicating that the discrepancy in WIMS-D is primarily related to the EDEP model. Choosing a more advanced energy deposition model in OpenMC with the EDEP 2 option brings the agreement with Serpent to within 200 pcm.

In the CORD-2 Krško NPP example, we observed that different nuclear data libraries significantly impact the predicted critical boron concentration, power defect, and reactivity gradient with burnup. ENDF/B-VII.1 provided the best prediction for cycle length, while newer libraries like ENDF/B-VIII.0 and JEFF-4T3 showed larger discrepancies. Specifically, JEFF-4T3 predicted a higher reactivity loss with burnup, leading to under-prediction of cycle length. The burnup gradient of the ENDF/B-VIII.1b3 test library was similar to the gradient calculated with the ENDF/B-VII.0 library. The study also identified that the over-prediction of power and xenon defects in ENDF/B libraries could be partly attributed to fission product yield data, with JEFF-4T3 performing better in this respect.

Our results emphasize that accurate depletion calculations rely on multiple factors beyond just cross sections. Energy released per fission, fission product yields, and the choice of nuclear data libraries are all critical components that influence the accuracy of reactor design calculations.

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Reactor physics

Parameterized cross-section library generation with the SERPENT code using ML methods and testing in VVER-1200 core geometry

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Calculation of the neutronics of a reactor core is usually undertaken in two steps, the first being the generation of assembly-wise group constants parameterized for various state parameters and burnup levels, and the second step is the core level calculation that utilizes these constants as input values.

In order to prepare these values for the second (core-level) code, a new parameterization module has been developed which uses various machine learning methodologies to be able to make accurate predictions in the phase-space of state parameters even when its training database is limited in size (as it happens when the group constants are generated with time-consuming Monte-Carlo codes).

This paper concerns first the generation of a cross-section library for VVER-1200 type assemblies using the new parametrization module and its implementation in the KIKO3DMG core-level nodal SPN code. In the course of the generation of the library, the homogenized assembly-wise group constants for the training database are generated for infinite assembly models by the Serpent Monte Carlo code for various burnup levels and state parameter values.

For the testing of the new library and the KIKO3DMG calculations, a full-core model of a VVER-1200-type reactor was set up. The reference solution was performed by the SERPENT Monte Carlo code for this core geometry and compared to the KIKO3DMG results for the infinite multiplication factor, the assembly-wise power distribution, reactivity coefficients, and control rod worths.

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Reactor physics

Application of Machine Learning techniques to the exploitation and interpolation of nuclear experimental data

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Machine learning (ML) approaches proved as powerful techniques already successfully applied to various fields such as weather forecast, languages, computer vision, agriculture, medicine, and others. The techniques use known information about some of the data to generate predicted properties for an unknown set. This work aims at investigating the possible use and performances of different ML methods for nuclear data evaluation and improvement and in different phases of experimental measurements including measurement configuration planning and post-analysis studies. Several application cases were studied.

The first case concerns the exploitation of the power distribution results measured at several location in the VENUS experimental reactor core. In the past, an extrapolation procedure, based on the Pattern recognition technique (RECOG-ORNL code cite{recog}) was already used in order to establish a complete 3-D map of the power distribution in the VENUS core. This procedure permitted at the same time to detect some suspicious or faulty values, transcription errors, as well as to give an idea of the accuracy of the neutron source interpolation procedure.

New ML methods were used this time to reevaluate the procedure and to deduce, from the available experimental data points, the complete power map in the reactor core and estimate the uncertainty in the

extrapolation procedure. Furthermore, as a key objective, the correlations between the measurements, and, in particular, the correlations among the corresponding uncertainties were estimated using the predictions of different fitting algorithms.

Different ML techniques were applied, such as Pattern recognition, Linear regression, Lasso regression, Ridge regression, Krigeage, Support Vector Machine, Random Forest and Neural networks. Predictions of different algorithms were compared to conclude on the method performances in terms of nominal values, average absolute and quadratic error, prediction coefficients, residues, and covariance matrices.

The outcomes will open the way to extend this study and apply the most promising method(s) to more general cases of measurements. A possible application is for measurement planning optimisation (experimental configuration, measured quantities - neutron spectra ranges, reaction rates, etc.) in the way to provide the experimental data needed for the improvement of the target nuclear data in the most efficient way.

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Reactor physics

Sensitivity analysis of the Krško NPP internal and external components to the activation of material impurities

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Nowadays, the decommissioning phase has to be planned already in the construction phase of a nuclear facility and the decommissioning plan has become part of the updated safety analysis report. Therefore, precise knowledge of the radioactive material produced in a nuclear facility is becoming increasingly important. We have used state-of-the-art particle transport methods in combination with activation methods to estimate the activity of the radioactive material of a typical 2-loop PWR. The calculation model is based on the geometric model of the pressure vessel and containment building of the Krško NPP. The Monte Carlo code MCNP was used for the simulation of particle transport and the FISPACT-II code for the activation calculations. The two codes were also used for a comprehensive sensitivity analysis. The aim was to evaluate not only the activities of the different reactor components, but also the effects of the different parameters on the calculated activities. Our study focused on the elemental composition of the materials and considered a wide range of impurities (over 40 in total) taking into account their minimum and maximum expected concentrations as specified in the NUREG CR-3474 reference report. Using site-specific neutron spectra and simulating 60 years of reactor operation, we calculated the maximum and minimum activities for core-level components such as the thermal shield (304 stainless steel), the pressure vessel (carbon steel), and the bioshield (consisting of concrete and rebar). While many parameters influence the overall activity, our analysis has highlighted the expected significant role of cobalt and europium contamination in determining the radiological impact during decommissioning. Our preliminary results show that both geometric variations or the presence of certain impurities can influence overall results by several 10 percent. Taken together, the combined presence of geometric defects and impurities can even lead to a difference of several hundred percent in the calculated activities. With this study, we aim to provide a possible indication of the sensitivity intervals due to material contamination for effective planning and safety assurance in decommissioning processes.

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Abstract Submissions

Assessment of component activation in small modular reactors

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The radioactivity of nuclear reactor's structural materials is induced by incident neutrons that are absorbed in the stable nuclei. Activated materials represent a significant part of radioactive waste in the process of nuclear power plant's decommissioning and their quantity has to be determined. In recent years, the development of nuclear reactors has shifted towards small modular designs, which would be fabricated in-shop with quick installation times on-site. However, the size difference generally implies a poorer neutron economy, meaning a larger portion of neutrons escape the system per unit power and interact with the structural materials, e.g. reactor pressure vessel.

In the present work we propose a method for assessing the activation of components in a SMR using the Monte Carlo neutron transport and activation analysis (FISPACT-II), specifically for the model of small modular reactor NuScale Power Module, a scaled-down pressurized water reactor, being the first SMR to obtain the design certification by the United States Nuclear Regulatory Comission (U.S. N.R.C.)[1].

[1] Emil Fridman, Yurii Bilodid, and Ville Valtavirta. Definition of the neutronics benchmark of the nuscale-like core. Nuclear Engineering and Technology, 55(10):3639-3647, 2023.

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Reactor physics

Adaptive Design and Criticality Analysis of the DARWIN Reactor Core Concept

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This paper addresses Charles Darwin's idea of adaptability and applies this concept to otherwise non-adaptable nuclear reactor. We introduce the concept of the Dispatchable Adaptive Reactor With Interchangeable CompoNents (DARWIN). Our goal is to develop an adaptable and versatile reactor design rather than optimising it for a single purpose. Such a reactor could fulfil specific needs and provide solutions for tasks such as pumping flood water, district heating, medical isotope production or desalination.

The first objective of this study is to outline the conditions required for specific applications, including medical isotope production and load following operation. The study presents the main physical characteristics of the current reactor designs available in the literature that should be integrated into DARWIN to increase its efficiency.

The second objective is to carry out an initial investigation of the effects of the fuel-to-moderator ratio on the physical characteristics of the reactor, particularly the effective multiplication factor (k-eff) and the power peaking factors, by changing the dimensions of the fuel cell. In this study, a hexagonal arrangement of the fuel pins within the assembly is assumed and the distance between the fuel pins in this hexagonal arrangement is systematically varied.

The calculations are performed in two dimensions with the Monte Carlo transport code Serpent 2, using only fuel assemblies with only one type of fuel without burnable poisons and without considering the entire reactor vessel. In the first phase of the study, UO_2 fuel with constant enrichment is used. Variations in the pitch between the fuel pins and the number of fuel elements in the core will be analysed to understand their impact on reactor performance.

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Reactor physics

Investigating the potential fuels for DFRm reactor concept

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Dual Fluid Reactor (DFR) is a concept design that combines the advantageous properties of two of the selected Gen IV reactor concepts; Molten Salt Reactor (MSR) and Lead Cooled Fast Reactor (LFR). In DFR molten salt is used (from MSR) in a separate circuit and is cooled by the molten lead (from LFR). In DFR, molten fuel flows inside the SiC fuel tubes while cooled by the lead that flows around these fuel tubes. The advantages of these two separate cycles enable the use of undiluted fuel salts and metallic fuels. Therefore, there are mainly two DFR concepts named DFRs and DFR_m which use molten salt and molten metallic fuel respectively. In this study, neutronic analysis has been conducted for the DFR_m design by using SERPENT code.

Reference DFR_m reactor uses U-Cr metallic fuel with the lowest temperature of 860 ˚C with % 4.78 wt. Cr, %12.8 wt%U-235 and 82.42 % wt. U-238 composition. To enlarge the operation temperature margin U-Ni and U-Fe fuel is proposed. U-Ni eutectic metallic fuel reaches its lowest melting point at 740 ˚C with 11 wt.% nickel and 89 wt.% uranium composition. As for U-Fe fuel, fuel composition at the eutectic point is %10.2 Fe and %89.2 uranium at 725 ˚C. It is shown that both U-Ni and U-Fe fuels have similar keff trends with the reference U-Cr fuel with lower k_{eff} values. This is because proposed fuel compositions have lower uranium content. To reach the same k_{eff} values U-235 content in the fuel must be higher for both U-Ni and U-Fe fuels. Thus, with these fuels in the DFR_m core, it is possible to reduce the risk fuel freezing risk and by this, the safety would be increased and ensure a wider operating temperature range.

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Reactor physics

Analyzing delayed neutrons in a pressurized water reactor

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Delayed neutrons are of crucial importance for the study of reactor physics, although they account for less than 1 % of the total neutrons produced during nuclear fission. They have a considerable influence on the dynamic behaviour of the nucleus. The majority of neutrons from nuclear fission are emitted almost immediately and are referred to as prompt neutrons, which come directly from the fission products. For analytical purposes, these precursors are usually categorised into six different groups, with the proportion of delayed neutrons in each group varying depending on the fissile material. The fraction of delayed neutrons (βi) for each group, is determined by weighting the values for the different isotopes based on the fraction of fissions and the region wise power within the core. Since the fuel in the core changes with burnup, the fractions of delayed neutrons are calculated for both the beginning and the end of the reactor's operating life. This paper presents a methodology for calculating the fraction of delayed neutrons and the lifetime of the prompt neutrons using the WIMS code. It also includes a comparison with the Monte Carlo code Serpent for a typical fuel cycle in the Krško nuclear power plant.

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Reactor physics

NuScale core analyses using Monte Carlo code Serpent

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Small modular reactors (SMRs) offer a more compact and economical alternative to conventional, larger nuclear reactors. As they're usually only a third of the size of conventional reactors, they require less space and can therefore be installed at a variety of sites. Various SMR concepts are being developed around the world, from light water reactors (LWRs), which use solid fuel and normal water for cooling, to boiling water reactors (BWRs), high temperature gas reactors (HTGRs) and others. NuScale's concept is one of the most advanced and the first to be certified by the U.S. Nuclear Regulatory Commission. NuScale's design is a pressurised water reactor (PWR) with a capacity of 160 MWth.

This work marks the beginning of a research initiative supported by the Slovenian Research and Innovation Agency (ARIS) and GEN energija, which started in 2023. The aim of the research is to investigate the impact of SMRs on waste generation. This first step presents the code-to-code comparison and includes 3D analyses of the entire core neutronics with a focus on integral core parameters and radial and axial power distribution, which will be compared with the results obtained by Fridman in their 2023 study on the neutronics of the NuScale.

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Reactor physics

Analysis of Power Mesh Tally Resolution for Coupled Neutronic and Thermohydraulic Simulations of Advanced PWR Core

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Coupled neutronic and thermohydraulics calculations are essential in designing advanced nuclear reactors. The increase in computational power in recent decades has fostered research in applying high-fidelity codes to investigate reactor cores. Data Exchange between coupled codes plays a significant role in the accuracy and convergence of coupled calculations. The present study focuses on analyses of power mesh tally resolution in Monte Carlo neutron transport simulation and its corresponding influence on fuel rod temperature distribution obtained from CFD calculation. A coarse mesh tally resolution requires a smaller number of simulated neutron histories; however, it leads to a coarse resolution of the power profile in the fuel, which influences the predicted temperature field. Neutron transport simulations are performed in Monte Carlo code Serpent, and CFD calculations are performed using the open-source software OpenFOAM. The influence of power mesh tally resolution is investigated on the 3D full-core simulation of the AP-1000 Pressurized Water Reactor, where two fuel assemblies were selected for the power mesh tally analysis. Further, the study includes an analysis of power distribution uncertainty in the selected fuel assemblies.

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Reactor physics

Optimizing Nuclide Sets for Efficient Monte Carlo Simulations in Large-Scale and Multiphysics Nuclear Reactor Models

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The paper presents an extensive evaluation of the influence of individual nuclides on key nuclear reactor parameters such as fuel reactivity, decay heat, and radiation sources, with the ultimate goal of identifying an optimized, reduced set of nuclides that maintain accuracy while enhancing computational efficiency in Monte Carlo simulations for large-scale nuclear models. Employing Monte Carlo simulation tools like MCNP or Serpent, which typically face challenges of slowed performance or increased memory demand when using extensive nuclide lists, this research investigates strategies to streamline these simulations without compromising essential details. Using the Teplator heavy water Small Modular Reactor (SMR) core as a detailed case study, the paper examines the effects of nuclide selection at various stages of the reactor cycle — Beginning of Cycle (BOC), Middle of Cycle (MOC) and End of Cycle (EOC). The paper assesses the trade-offs between simulation detail and computational demands by studying deterministic and Monte Carlo calculation approaches. The findings demonstrate the feasibility of significantly reducing the number of nuclides in simulations and highlight how such reductions can be implemented in practice, ensuring that computational resources are judiciously used while retaining a high degree of accuracy in predicting reactor behavior throughout its lifecycle. This approach offers a practical pathway for enhancing the Monte Carlo simulations' efficiency in designing and analyzing large, complex, three-dimensional nuclear reactor models or multiphysics simulations coupling neutronics and thermal hydraulics.

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Reactor physics

Liquid neutron filter for experimental simulation of the thermal neutron spectrum shift to higher energies

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A new experimental device, referred to as the liquid filter, is currently under development to simulate the thermal neutron spectrum shift to higher energies. The device will comprise two planar fission cells with differing fissile materials for comparative analysis, encased within borated water. The addition of boric acid to water serves a dual purpose: it lowers the thermal peak in the neutron spectrum while concurrently shifting it to higher energy levels, thereby facilitating measurements of cross sections important for temperature feedback effects.

This paper delineates the design process of the device alongside preliminary calculations. Our primary aim was to ascertain the optimal design parameters including size, materials, and geometry, in addition to evaluating its impact on the criticality of the IJS TRIGA reactor. Furthermore, we conducted calculations on the response of the fission cells.

The calculations were executed using the MCNP code. To streamline the process of finding the optimal design and conducting preliminary calculations, we utilized a simplified planar neutron source. Subsequently, when

integrating the device into the IJS TRIGA model, ADVANTG was employed for neutron spectrum and detector response calculations, thereby minimizing variances and computational time.

The impact on the TRIGA reactor's criticality was found to be minimal. The most suitable measuring location is yet to be determined. Possible options include radial, tangential, and thermal column channels. The selection will be based on the fission cell response results. It is imperative that the neutron spectrum and, preferably, neutron intensity remain consistent at both fission cell locations, although adjustments may be made based on the experimental methodology.

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Reactor physics

Reactivity effects of external void insertion in Lead-cooled Fast Reactor

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The utilization of liquid lead as a coolant within the active zone of the ALFRED reactor offers additional reactivity control possibilities, leveraging neutron leakage phenomena. With its high density and atomic mass, lead effectively serves as a neutron shielding; thus, manipulating its displacement allows for modulation of the neutron population within the core, enabling adjustments to reactivity levels, especially close to the core boundary.

In this study, we investigate the possibility of employment of void insertion, which is applied within the channels of the reactor's safety system for reactivity control purposes. Utilizing the OpenMC Monte Carlo neutron transport simulation code, we analyze the effects caused by void insertions in the channels of III and VII rings of the reactor core's assemblies. Our analysis unveils non-linear trends in the relationship between reactivity insertion and introduced void volume. Furthermore, this work explicates the obtained results and considers the potential applications of void insertion.

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Reactor physics

Validation of the MCNP and OpenMC Monte Carlo Codes for the Nuclear Criticality Safety Calculations of the NPP Krško Fuel

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In the criticality safety calculations, adequacy of the applied safety margin has to be demonstrated. An appropriate bias should be determined from the difference between calculated and experimental results, reflecting the accuracy of the used calculational methodology. The bias and the uncertainty associated with the bias are used in combination with an additional subcritical margin to establish an Upper Safety Limit (USL). An adequate subcritical margin is considered to be ensured, if the calculated results are below the USL and within the area of applicability of the code validation.

In order to validate the MCNP and OpenMC codes for the criticality calculation of the NPP Krško fuel, we identified ten benchmark experiments from the International Handbook of Evaluated Criticality Safety Benchmark Experiments, also known as the ICSBEP handbook. The cases were selected from the Low Enriched Uranium (LEU) and Mixed Plutonium – Uranium (MIX) sections to cover fresh and irradiated fuel.

In addition to the determination of the specific USLs, several additional analyzes were performed. An impact of the nuclear data was investigated by comparing the results obtained with the ENDF/B-VII.1 and ENDF/B-VIII.0 nuclear cross sections. The OpenMC code has recently become quite popular as it is an open Monte Carlo code that does not require a special license. Therefore, it was decided to include it in the project to explore its strengths and weaknesses by comparison to the well-established and widely used MCNP code.

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Reactor physics

Numerical investigation of Yttrium-Hydride Reactivity Feedback Coefficient

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Yttrium hydride (YH_x) is an appealing candidate as advanced moderator for micro and space fission reactors due to its strong neutron moderation capability. This allows for compact core designs and relatively high operating temperature if compared to other alternatives such as zirconium hydride. Despite YH_x promising features, its use still requires numerous analysis for characterising its thermomechanical and neutronic properties.

This works aims at providing a better understanding of the YH_x moderator temperature feedback coefficient (MTC) from a neutronic point of view. To do so, various geometrical configurations has been taken into account and modeled in OpenMC monte carlo code. These are: fuel pin surrounded by moderator, fuel and moderator slabs and finally moderator pin surrounded by fuel. For each, the MTC has been evaluated and decomposed in its main contributions, namely, the thermal utilisation factor, the resonance escape probability, the fast fission factor and the reproduction factor. Because of the upscattering phenomena which happen in thermal energy region due to crystal and molecular structures, the analysis of temperature feedback coefficients necessitates the use of appropriate scattering laws. Specifically, for YH_2, these laws are accessible within the ENDF/B-VIII package, hence these libraries were chosen for this study.

The analysis shows that the MTC is positive but decrease at higher moderator temperature, regardless the geometrical configuration. Moreover, moderator-to-fuel volumes ratio strongly affects MTC value. Among the aforementioned contributions, the thermal utilisation factor is the most relevant. The decomposition of the 4-factors formula revealed that the main reason for the positive feedback coefficient is the change in the ratio between neutron absorption in the fuel and in the moderator caused by neutron spectrum hardening.

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Reactor physics

Preliminary Monte Carlo calculations for ex-vessel neutron dosimetry gradient chains at Krško NPP

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Many nuclear power plants (NPPs) in Europe are aging, requiring an assessment of the state of embrittlement of the reactor pressure vessel, as well as concrete degradation, swelling of internal components, internal corrosion cracking, and other problems. During the 25th operating cycle of the reactor, an Ex-Vessel Neutron Dosimetry (EVND) program was implemented at the Krško NPP to continuously monitor the reactor pressure vessel beltline. As part of this program, three sets of stainless steel gradient chains were placed at azimuthal positions corresponding to 0°, 15° and 30°. These neutron dosimeters are suspended in the annular space between the reflective insulation of the reactor pressure vessel and the concrete biological shield, at a radius of approximately 197.65 cm from the centerline of the reactor core. At all three azimuthal positions, the active sections of the stainless steel gradient chains extend approximately 2 meters above and below the mid-plane of the core. These segmented chains facilitate the measurement of iron, nickel and cobalt reactions ($^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,p)^{58}Co$ and $^{59}Co(n,\gamma)^{60}Co$), which are used to determine the axial and azimuthal gradients. The EVND creates a facility-specific database that facilitates the assessment of vessel exposure and the uncertainty of this exposure throughout the operational lifetime of the facility. Furthermore, those measurement are important for the absolute validation of the ex-vessel neutron transport calculations with newly developed computer codes and models. The Monte Carlo neutron transport code MCNP was used to model the containment building of the Krško NPP in conjunction with a previously verified and validated MCNP core model to predict the measured reaction rates of the EVND gradient chains.

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Reactor physics

Computational analysis of a neutron diode

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The use of neutrons for wireless communication was demonstrated in 2022. A computational study of the transmission of a neutron pulse through various materials and the possibility of signal amplification was carried out, which showed promising results for the development of neutron communication. These discoveries open the potential for the use of neutrons in computational circuits. For the latter, it is necessary to obtain diode elements, i.e. an element with asymmetry in the neutron transmission with respect to the incident direction. The aim of this computational study was to design a combination of layers of different materials that would have the highest possible ratio in the transmission of fast neutrons between forward and reverse directions. Diodes were modelled as a one-dimensional objects consisting of two layers – a layer of absorber and a layer of moderator of differing widths. The moderators used in the study were water, beryllium, heavy water and polypropylene, while the absorbers we used were natural cadmium, boron carbide, uranium-238 and Ag-Cd-In alloy. In this paper, the results of the simulations carried out with Monte Carlo particle transport code MCNP 6.2 code and variance reduction technique FWCADIS using ADVANTG v3.2 will be presented. In addition to the efficiency of diodes, other important features for the implementation of the diodes into computational circuits, such as the attenuation of the neutron flux and the shape of the neutron pulse, will be presented.

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Reactor physics

Sampling cross sections from the nuclear data libraries

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The results of simulations are only useful within the uncertainty. The uncertainty of calculations depends on the uncertainty of the input parameters, the correlations between them and the response function. For simple, linear systems with known sensitivities the uncertainty can be calculated analytically, using the sandwich formula. When our systems have complex dependencies between parameters or nonlinear behavior, sampling methods can be used. Another benefit of the sampling methods is that they can handle arbitrary distributions of input values. The covariance information for the incident neutron reaction cross sections is stored in file 33 and file 31 for nu-bar. The challenge is that covariance matrices are not measurable physical quantities, but rather a mathematical description of the uncertainties and relations between the parameters. They depend on the

evaluation methodology and the data taken into account. This means that the correlation matrix cannot be tested for validity, except indirectly by judging the magnitude of the calculated uncertainties in simulations compared to measured values in integral experiments. There are, however, certain mathematical properties, such as positive definiteness and symmetry that the covariance matrix needs to fulfill to be considered correct. We have used NJOY to process the covariance information from the ENDF files and an open-source program Sandy developed by Luca Fiorito to draw random samples based on the covariance matrix. We have checked the processed covariance data for mathematical inconsistencies and verified how these issues impact the sampling procedure. In this contribution we are presenting an analysis of the covariance data in different nuclear data libraries, check for mathematical correctness and comment how the issues impact the sampling performance.

Research reactors

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Research reactors

The use of low power research reactors to develop new protocols in the pursuit of exotic radionuclide production

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The field of medical radionuclide production is well established, sinking its roots in the possibilities given by nuclear reactors and cyclotrons alike. Nonetheless, there is a constant drive for the development of new and exciting protocols leading to the discovery of novel radionuclides with vast applications.

It is in this setting that Pavia's TRIGA MK II research reactor fulfills its purpose, where the synergic collaboration with the University's departments, allows for the upbringing of a new wave of bachelor's, researchers, and doctorates in the domain of nuclear sciences, which in turn grants quality publications and efficient use of the facility both in terms of research progress and in meeting LENA's training goals. The apparent limit of low-power reactors with insufficient fluxes for the commercial production of commonly employed radionuclides can be mitigated by switching the focus to a research and training approach. This led to the production and separation of various exotic radioisotopes in the form of ^{111}Ag, ^{161}Tb, ^{64}Cu, ^{18}F, ^{166}Ho, and ^{99m}Tc through the Neutron Activation process, and fruitful collaborations with many organizations such as INFN for the ISOLPHARM project, and ENEEP that have helped to highlight the potential of this type of facilities and the possible contributions that they can bring to life.

This work will focus on the milestones of the LENA facility in terms of training and research with particular attention on data gathered over the years.

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Research reactors

Review of SiC based sensor performances for neutron flux measurements in nuclear reactor fission facilities

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Silicon Carbide (SiC) based neutron sensors present very promising properties for neutron flux measurements: compactness, radiation and temperature hardness, selectivity between neutron and gamma radiations, response linearity, pixelation… In particular, p-n and Schotcky diode designs are studied since more than 20 years for almost all fission, fusion, medical, spatial and high-energy application cases and applied for some of them [1]. CEA IRESNE and AMU/IN2MP French research organisms also started in 2012 the development of a SiC based sensors for nuclear fission research reactors, and in particular for a use in intense and variable neutron flux, wide neutron spectrum energy and mixed gamma and neutron radiation field, such as ones encountered in Material Testing Reactor. Temperature influence and methods for deriving absolute neutron flux level for different neutron energies have also been studied.

This article begins with an updated review of the various results available from literature with the aim of an application for measurements in nuclear fission reactors. The analysis of these results then enables us to identify the main performances achievable by these sensors in relation to the targeted applications (thermal or fast neutron flux measurements, for example). It also leads to identify some limitations and to propose possible ways of improvements, such as, for example nuclear data, as conclusion.

[1] F. Ruddy et al. “Performance and Applications of Silicon Carbide Neutron Detectors in Harsh Nuclear Environments” - ANIMMA 2021 - EPJ Web of Conferences 253, 11003 (2021) https://doi.org/10.1051/epjconf/202125311003

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Research reactors

Shadowing Effect Correction for the Pavia TRIGA Reactor Using Monte Carlo Data and Reduced Modelling Techniques

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Reactor safety and monitoring have historically been important and challenging aspects within nuclear power plants, and attention to these issues will grow even more in future years due to both the increase in the energy demand, including from fission plants, and the unique features of designs expected for Gen-IV reactors. Neutron flux measurement techniques play a crucial role in both safety and efficiency in nominal reactor operations situations, being the flux a quantity directly correlated with the overall plant power and subjected to faster dynamics compared to thermal-hydraulics quantities. However, as is well known in the literature, the neutron flux undergoes spatial distorsions due to the insertion of the control rods. This in turn can result in the shadowing of detectors, interacting with the measurements of significant reactor parameters from the point of view of safety, corrupting experimental data. These effects are even more pronounced in small reactors, such as research facilities or micro-reactors, where few high reactivity worth control rods are used. For this reason, this paper focuses on the TRIGA Mark II research reactor at the University of Pavia, proposing the calculation of a correction factor for the shadowing effect applied to the flux estimation. The correction factor is inferred using Reduced Order Modelling techniques applied to parametric OpenMC simulations representing several nominal TRIGA configurations.

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Research reactors

Enhancing Irradiation Facilities: A Study on the Upgraded Water Filtration System in the KATANA Water Activation Loop

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In 2023, the experimental operation of the KATANA water activation facility began in the hall of the TRIGA Mark II nuclear reactor at the “Jožef Stefan” Institute (JSI), which was designed and built for research into water activation under fusion-relevant conditions. The TRIGA reactor at JSI can be operated in both steady-state and pulsed mode. This enables continuous measurements on the KATANA device over several hours and the possibility to simulate the operating regime of fusion reactors. KATANA will provide detailed insights and the possibility to validate calculation methods for the water activation process under fusion-relevant conditions and much more, which is directly relevant for the preparations for the operation of the large fusion research reactor ITER.

During the test phase of the KATANA water activation loop, extensive operational experience was gained, leading to insightful feedback on its performance and functionality. Based on this comprehensive evaluation, several important improvements were proposed to increase the efficient-accurate control of operating conditions within the loop. These proposed improvements aim to enhance the overall performance of the KATANA water

activation loop, making it safer, more effective and easier to operate in various settings. This article details the design, construction and operational aspects of the water filtration system within the KATANA water activation loop. The filtration system is designed to remove contaminants and keep the water as pure as possible to reduce experimental background. To evaluate the effectiveness of this system, water conductivity measurements were taken before and during the filtration process. These assessments were essential in confirming the system's ability to effectively remove contaminants and improve water purity. In addition to the conductivity measurements, the study also included a comparative analysis of gamma spectra during the operation of the loop, taken before and after the water was filtered. This comparison was crucial for understanding the radiological properties of the water before and after filtration. By analysing the gamma spectra, insights were gained into the types of radioactive isotopes present and the effectiveness of the filtration system in reducing water contamination. This dual approach provides a comprehensive overview of the performance of the filtration system within the KATANA water activation loop.

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Research reactors

Jožef Stefan Institute TRIGA Research Reactor Activities in the Period from September 2023 – August 2024

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The Jožef Stefan Institute (JSI) has maintained operational oversight of a 250 kW TRIGA research reactor since 1966. Over the past decade, continuous monitoring of Safety Performance Indicators (SPI) has been conducted, encompassing critical parameters such as operating time, the number of irradiated samples, doses received by operating staff, and the activity of radioactive gases released into the environment. This paper provides an in-depth analysis of SPIs observed throughout the year 2023, which is essential for enhancing the reactor's future operational safety.

In the field of research, we have continued most of the established research campaigns from previous years. Notably, numerous samples were irradiated for CERN and associated partners within the EURO-LABS initiative. Collaborative efforts with CEA yielded significant advancements, particularly in enhancing passive neutron and gamma-sensitive detector performance. Additionally, participation in the TIFANY (Tellurium for Epithermal Neutron Dosimetry) project involved the irradiation of Tellurium and tin samples. In June, we plan to test beryllium SPND (Self-Powered Neutron Detectors) for fast neutron detection. In June, we also plan to test radiation hardness for several types of glass which could be used for optical fission chambers.

A novel experimental device, the Water Activation Loop situated within beam port no. 1, is set to be formally commissioned in June. Demineralised water gets activated in the vicinity of the core. The water is led inside the shielded locations in the reactor hall, which can be a gamma radiation source of 6 MeV and 7 MeV rays. The preliminary results will be presented, including the achieved dose rate at measuring locations during steady state and pulse mode operation.

In the field of education, many exercises were performed by students from the following universities: University of Ljubljana, Uppsala University, Aix Marseille University, and Politecnico di Milano. Besides that, one EERRI and two ENEEP courses were organised where participants were young professionals in the nuclear field. For the first time, two Nuclear technology courses were organised in the same year. The first was attended by NEK personnel, and the second was attended by Gen Energija personnel.

In 2022, all three thermocouples failed in one instrumented fuel element. Activities were started to obtain an instrumented fuel element. The paper will present several options.

Innovative solutions, such as a new heating system leveraging residual heat from the computer cluster's cooling system, were implemented during the winter season. Further, a new radiation monitoring system was deployed in the summer of 2024.

Significant progress was achieved within the operational team, with the recruitment of a new operator who is anticipated to obtain an operating license in the coming year.

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Potential effect of saline solution on fission products mitigation by pool scrubbing

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The potential release of fission products (FPs) to the atmosphere during severe nuclear power plants accidents is a main issue in nuclear safety. Accurate estimation of these releases is crucial for risk assessment and the implementation of appropriate measures. This study focuses on the process of "pool scrubbing", specifically the mitigation of FPs transported in a carrier gas injected through a liquid pool. Pool scrubbing can occur in various accidental situations in PWRs (Pressurised Water Reactors), including accident sequences leading to the use of FCVS (Filtered Containment Venting System) and SGTR (Steam Generator Tubes Rupture), as well as in nuclear powered submarines where the proposed solution is to submerge the vessel. It is also worth noting that pool scrubbing scenarios can occur in Small Modular Reactors (SMRs) using pressurised water, such as the European NUWARD project.

A dedicated facility named TYFON (Trapping and hYdrodynamic for FissiON behaviour in pool scrubbing) at IRSN enables to study the bubble hydrodynamic behaviour mainly near the nozzle and the retention of iodine compounds for different carrier gas injection flow rates, and thus for several Weber numbers. This latter dimensionless number reflects the effect of the inertia against the capillary forces. One of the assets of this facility is that we can study both retention of iodine compounds and hydrodynamic behaviour of bubbles but only near the injector. In order to complete the study of the pool scrubbing phenomena, the hydrodynamic of the plume is studied with another facility, named BULLET (BUbble Liquid flow: pLumE sTudy). The objective is to provide a comprehensive characterisation of the full flow in both clear and salt water within this specific zone.

One of the main contributions of this study is to characterize the hydrodynamic of bubbles and the trapping of iodine compounds in a saline environment, with a particular focus on volatile iodine (I_2 and CH_3I) and aerosols (CsI, CsOH…). Indeed, a very little data can be found in the literature in such environment. The aim is also to improve more generally the characterisation and modelling of the hydrodynamic of bubble plume formed at a distance from the nozzle.

Indeed, a previous work was focused on the zone of injection, on TYFON facility, where the characteristic bubble structure (volume, surface) near the injector was obtained thanks to image processing. The bubble volume is quantified from images obtained by a high-speed camera operating at 3.5 kHz. To ensure the highest degree of accuracy, an axisymmetric assumption is made. This methodology has been validated and highlighted in Farhat et al. *(Characterization of bubbles dynamics in aperiodic formation, International Journal of Heat and Mass Transfer 180, 2021*). Also, the evaluation of the decontamination factors (DF) in clear water were conducted in different regimes, from bubbling to jetting, notably for CsI aerosols *(Farhat et al., Hydrodynamic aspects of aerosols pool scrubbing, Chemical Engineering Research and Design 191, 2023)*.

In the present study, preliminary literature review was carried out, examining the hydrodynamic and chemical behaviour of the retention of iodine FPs in saline solution. For the hydrodynamic part, the literature results highlight the influence of electrolytes on surface tension and bubble size in a saline context. Furthermore, it is commonly observed that the presence of salts leads to a decrease in bubble velocity due to transport phenomena at their surfaces, called the Marangoni effect *(Firouzi et al., A quantitative review of the transition salt concentration for inhibiting bubble coalescence. Advances in Colloid and Interface Science*).

Plus, a decrease has been noticed in the bubble size caused by coalescence inhibition at a specific critical salt concentration *(Craig et al., Effect of electrolytes on bubble coalescence. Nature*).

Regarding iodine retention, there is very few results in the literature about the effect of electrolytes on the mitigation of these FPs. Initial retention tests of iodine species were conducted at IRSN in 2021 in a saline environment showing great mitigation of I_2.

For the current study, preliminary qualification tests were done to optimize conditions for future tests in a saline environment. To achieve this, a comparison is made between the results (hydrodynamics and retention) obtained with prototypical seawater according to most accurate norm in the literature, ASTM D1141 and water containing only Sodium Chloride, which constitute about 60%w of the total amount of salt in seawater. The aforementioned tests were performed on TYFON device. First results showed a significant impact of the presence of salt in the retention of iodine. Indeed, great DFs were obtained with just the presence of sodium chloride for I_2. Nevertheless, no discernible difference was observed in the bubble volume at the injector between seawater, water with sodium chloride and clear water.

The experimental data collected, including hydrodynamics and DF, will be used to improve and to validate the modelling of the pool scrubbing phenomenon in the ASTEC integral code (Accident Source Term Evaluation Code), developed at IRSN.

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Application of system and CFD codes on simulation of PERSEO experiment passive heat removal systém within the EU PASTELS project

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The EU PASTELS project aims at demonstration of how innovative passive safety systems can support modernisation and optimisation of the European nuclear industry. Within the project, several passive heat removal facilities comprising of single, combined and integral tests have been studied both with system and CFD codes. One of these facilities is PERSEO (In-Pool Energy Removal System for Emergency Operation) integral facility, which was built at SIET laboratories, modifying the existing PANTHERS IC-PCC facility (Performance Analysis and Testing of Heat Removal System Isolation Condenser - Passive Containment Condenser), used in the past for testing a full-scale module of the GE-SBWR in-pool heat exchanger. The scope of this paper is to summarize the efforts conducted by the members of the PASTELS consortium on the PERSEO Test 7 and to evaluate the applicability of the CFD and system codes on simulating the passive heat removal systems, reveal insufficiencies in modelling and to create a cornerstone for future development and benchmarking activities.

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Thermal-hydraulics, computational fluid dynamics

Exploring Taylor Bubble Dynamics in Counter-Current Flows: A Combined Numerical and Experimental Study

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Fluid dynamics is a fundamental pillar of modern engineering, playing a pivotal role in enhancing the efficiency and safety of various industrial and energy systems. In nuclear power systems, understanding two-phase flows becomes particularly critical during events such as boiling heat transfer and accident management scenarios, including the Loss of Coolant Accident (LOCA). During these scenarios, the slug flow regime—characterized notably by elongated, bullet-shaped bubbles known as Taylor bubbles—often occurs, particularly within steam generators. This can pose significant challenges to the stability of the system.

This study focuses on the behavior of Taylor bubbles in counter-current air-water flows through a dual methodology approach, incorporating both high-fidelity numerical simulations and advanced experimental techniques. We observed the dynamics across two distinct flow regimes: a transitional flow with Reynolds number (Re) of 1400 and a fully turbulent flow with Re of 5600. The investigation focused on bubbles of varying lengths under stagnant conditions, where buoyancy is balanced by the downward flow's inertial drag.

Experimentally, we concentrated on three pivotal aspects: the disintegration of bubbles, in-depth analysis of bubble interface dynamics, and velocity field measurements in the liquid phase with Particle Image Velocimetry technique. High-speed video techniques and in-house image analysis algorithms revealed bullet-train-like asymmetries in the Taylor bubbles, challenging the anticipated axisymmetric profiles characteristic of co-current Taylor bubbles. Disturbance waves on the bubble interface were also detected, whose correlated motions across the surface provide new insights into bubble behavior in such flow conditions.

Numerically, we used the open-source CFD framework OpenFOAM to study bubble behavior and disintegration mechanisms, employing a high-order Runge-Kutta time integration schemes alongside a Volume-Of-Fluid (VOF) approach for precise bubble interface reconstruction. Our analysis primarily addressed the transitional flow regime with a focus on the efficacy of algebraic versus geometric interface capturing techniques. Notably, we observed the formation of a secondary vortex in the turbulent wake at finer mesh resolutions. Countering the bubble breakup, a new grid-scale surface tension model has been proposed.

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Design and construction of the transparent test section for analysis of boiling flow scaled to fusion divertor conditions

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The new transparent test section scaled to fusion reactor conditions has been constructed. To scale-down the extreme conditions of high heat flux and high velocity from the divertor water-cooling channels to laboratory conditions, a similarity analysis has been performed. The relevant similarity numbers (Re, Bo, We) have been used to transform the key operating parameters (pressure, mass flux, enthalpy, heat flux) and select a surrogate working fluid R245fa that allows the experiment to operate at much lower heating power and pressure appropriate for materials of a transparent test section. By maintaining similar non-dimensional numbers, the water to refrigerant scaling aims to preserve the geometric, hydrodynamic and thermodynamic similarities of the cooling channels.

The test section is designed as a transparent horizontal square duct with a cross-sectional width (10 mm) approximately similar to the size of the water-cooled divertor channel. The test section is made mostly out of transparent polycarbonate, which enables visual observation and recordings by a high-speed (HS) camera. To replicate the heat load conditions in the cooling channel, heating from the top is provided by an electrically heated thin stainless steel foil, directly in contact with the fluid. The foil is mounted underneath an infrared (IR) transparent window to allow IR measurements. The foil is painted black to ensure high accuracy of IR thermography. IR camera will be used to measure the surface temperature on the heated foil, while HS cameras in visual light will be used to visualize convective boiling flow from the side and from below. In this study the design and construction of the new test section is described in detail.

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Lagrangian simulation of flow boiling experiments in horizontal annulus

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Flow boiling is an important heat transfer process in many engineering applications, most notably in nuclear power plants. The prediction of flow boiling phenomena is crucial for efficient and safe operation, but remains challenging to this day, therefore numerical simulations are commonly used for this task. Most commonly, single-phase and two-phase flows are simulated using the Eulerian approach by discretizing the computational domain and solving a set of basic fluid dynamics equations. The Eulerian approach uses time averaging to solve two-phase flow fields in a fixed observation volume. This method enables efficient solution of complex flows with a large number of bubbles, but only indirectly models interactions between bubbles. Lagrangian method on the other hand, is a particle-based approach that can track the motion of individual bubbles or fluid particles and models their interactions directly. With a known liquid velocity field and a sufficiently small number of bubbles, such a simulation can reproduce the real motion and interactions between bubbles and can be computationally more efficient, enabling a near real-time calculation. In our study, a Lagrangian simulation of boiling flow experiments with different bubble behavior was developed. The basic equations, computational principle and qualitative comparison with experimental results is presented.

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Noncondensable gas – Liquid interface issue in LFRs: investigation through the manometer test case

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In the last years, the increase of energy demand and the need of sustainable energy production have newly focused the interests on liquid metal cooled fast reactors (LMFR).

The lead-cooled fast reactor (LFR) represents a possible solution because of its excellent compatibility with a fast neutron spectrum, chemical and thermal-hydraulic characteristics safety oriented. However, the experience gained in operating such reactors is very limited, as far as experimental facilities covering the targeted operating conditions are rare and, therefore, the role of simulations becomes crucial for the design and the safety analysis.

System codes are widely used for simulating reactors' behaviour in the steady state and transient conditions, because of their reliability in catching the main phenomena involved in existing nuclear power plants. The extension of their application to Liquid Metal cooled Reactors is possible and some examples are proposed by the international community.

In spite of this, some important issues are still open and, among them, the simulation of a liquid and a non-condensable-gas interface is of particular interest because of its influence on reactor operational transient simulation's correctness.

This interface and its behaviour during transient are analysed in this paper, taking as test case a manometer filled partially with liquid metal and the rest with a non-condensable gas. The feature of the problem is characterized by parametric analyses that might influence the results and arise from geometry and boundary conditions in the simulations.

A benchmark has been performed among different system codes, such as RELAP5/Mod3.3, ASYST_LM and ATHLET and the obtained results have been compared also with the analytical solution. The outcome has evidenced different codes' capability to afford the problem and possible improvements which are mainly connected to the calculation scheme adopted and which present a status of the level of maturity of such codes for LFR simulations.

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Influence of Chemical and Physical Treatments of Copper Surface on the Contact Angle of R-245fa Refrigerant

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Efficient heat transfer is crucial in nuclear power plants as it facilitates the optimal heat removal from the reactor core and enhances performance of steam generators as well as condensers. A significant effect on boiling heat transfer properties has a contact angle or wettability of a solid surface by a liquid. In this study, we investigated the influence of various chemical and surface roughness treatments on the contact angle of refrigerant R-245fa on cylindric copper surface, which is relevant for our flow boiling experiment in the THELMA laboratory at Reactor engineering Division, Jožef Stefan Institute. Surface roughness of copper was modified with sandpapers of different grit sizes applied in axial and tangential direction, by electrodeposition, sandblasting and etching. Chemical modifications included teflon (PTFE) and graphene treatment. The contact angle measurements have been performed from images captured with a microscope. The measuring method has been validated by comparing the contact angle measurements of water on the same surfaces with the data from existing literature. The findings indicate that altering surface treatments, like changing chemical properties and roughness, affects how R-245fa behaves. When we made surfaces rougher on micro and nano scales, the contact angle increased. Using chemicals that reduce surface energy of solid phase also helped increase the contact angle. Combining roughness and chemical treatment has the biggest effect on increasing the contact angle.

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The CHIP Line: an unique experimental device dedicated to fission product chemistry and transport at high temperature in PWR severe accident conditions

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The CHIP experimental line developed at IRSN (Institut de Radioprotection et de Sûreté Nucléaire) allowed since 2008 to perform experimental research on the behaviour of fission products (FP) in relation to severe accidental nuclear reactor situations. Particular attention has been paid to iodine and caesium, which are the two major contributors to the short- and medium-term radiological consequences in the event of significant outside airborne releases. The aim is to understand as finely as possible their physico-chemical behaviour from the release of degraded fuel to the containment. The final goal is to refine/validate the ASTEC simulation tool developed by IRSN, which make it possible to assess potential releases to the environment – the so-called source term.

If numerous information on releases from the degraded core could be gained from the integral PHébus FP tests, some important issues were pointed out - in particular concerning fission products reactivity during their transport in the reactor coolant system (RCS) [1;2]. In order to better understand the main chemical formation/decomposition/deposition processes in which released FPs and control rod elements are involved in the RCS during a SA and also to build an experimental database allowing the validation/development processes for models, the CHIP experimental set-up (CHemistry Iodine Primary Circuit) has been developed over the past decade within the framework of the ISTP (International Source term Programme) - CHIP (2005-2012) programme and through the CHIP+ (2012-2018) bilateral programme between IRSN and EDF. The CHIP set-up is dedicated to the study of complex systems involving up to 8 elements in its last configuration. It is designed as an open flow reactor in which the reagents are continuously injected and mixed at high temperature (1600°C) and then transported into a controlled thermal profile and carrier gas (steam/H_2) simulating as much as possible the conditions of the RCS of a Light Water Reactor (LWR) during a SA. Deposited species, transported aerosols and gases are collected for off-line characterisation with a focus on the detection and quantification of gaseous iodine which may be released at the cold break. Chemical systems involving the main released FPs (I, Cs, Mo, Te) and control rod material (B, Cd, Ag and In) were considered with increasing complexity. One of the main issues was to achieve element molar ratio covering conditions of interest to highlight possible chemical reactivity.

After ten years of work performed within the framework of the CHIP and CHIP+ programmes, with large efforts in order to develop/qualify the experimental facilities as well as the controlled elements injection and to identify and validate the appropriate chemical analysis, the CHIP experiments brought some key elements of understanding to better simulate this accident phase. It was indeed confirmed that beside CsI, other iodine forms (gaseous /condensed) can be transported in the RCS in severe accident conditions [3]. The persistence of a significant gaseous iodine fraction in cold leg break conditions seems to be strongly linked to nature of gas atmosphere and the ratio of the other released elements – especially Mo/Cs ratio [4]. Control rod elements (Boron for B4C and Ag, Cd for SIC control rod) were highlighted as playing also a role on FPs transport. Efforts are now ongoing to improve modelling in considering for instance non congruent processes of condensation as well as completing the thermodynamic database.

More recently, within the frame of the OECD ESTER program (2020-2025), the CHIP facility was renewed to allow for experiments involving the study of revaporisation phenomena and thus better qualifying medium term releases as highlighted during the Fukushima Daichii accident (2011). Semi integral tests allowing to assess potential delayed FP releases are presently on going. Short term perspective will be the study of the role of Chromium on FP chemistry in SA condition. Indeed The new nuclear fuels known as ATF (Accident Tolerant Fuel) based on chromium-doped uranium oxide/ or Cr coated claddings have interesting properties with better resistance to oxidation in the event of loss-of-coolant accident (LOCA) leading to a moderate heating of the fuel and less release of fission products. On the other hand, under severe accident conditions with partial fuel meltdown, this type of fuel has been much less studied and the question of chromium reactivity with the other main FPs is raised. This future work will be a part of the FORESEEN OECD technical proposal which could succeed to ESTER project.

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Safety analyses and severe accidents and PSA

Decay heat removal from the MSFR core through the passive safety system

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Molten Salt Reactor (MSR) concept was initially developed at Oak Ridge National Laboratory, and the first Molten Salt Reactor Experiment (MSRE) went critical already in 1965. Currently, MSR concept again receives an increased interest from researchers. In particular, the European design, Molten-Salt Fast Reactor (MSFR) was studied in framework of the SAMOSAFER project, following SAMOFAR, EVOL and other European projects.

Current licensing requirements impose a need to design proper safety systems which can either prevent an accident, or mitigate its consequences. All generation IV reactors are required to be equipped with passive safety features. A postulated accident in MSFR is loss of forced flow due to a pump trip. In the past, during loss-of-flow accident, it was assumed that the fuel salt has to be discharged to the emergency draining tank and cool down there, by passive, natural-circulation, cooling loops [1,2]. In this paper, we study a new passive safety feature. In this scenario, fuel salt remains in the core and proposed modifications in the design should provide enough cooling to safely turn off the reactor, under an assumption of subcriticality. In this modification, the emergency draining tank remains in its place, adding an additional layer of defence in depth. After the primary circuit circulation pump is stopped, the flow in the primary circuit is maintained by natural circulation, enhanced by temporary increase of temperature in the core. In the same time, negative feedback coefficients promptly bring power down to 5% within a second [3]. The heating power remaining in the fuel salt causes the material to heat-up and forces an upward motion of the hotter fluid. The hot fluid reaches the intermediate heat exchanger (IHX) located at the top of the primary loop and transfers heat to the secondary (inert-salt) loop, that also operates on natural circulation basis only. Significant height of this loop provides sufficient natural circulation for further heat transfer to the tertiary loop. The tertiary loop can contain inert salt characterized by a low melting point, or water. Finally, the decay heat has to be rejected to the environment.

In this study, the in-core and primary loop parameters are calculated by the SIMMER-III code, which is a mechanistic tool for complex evaluation of severe accidents. It is a 2D fluid-dynamics code coupled with a structure module and neutron transport code. In support of the study, three small code modules were developed to quantify the heat transfer in the heat exchangers. At the moment, two modules are used iteratively to find the flow rate in each circuit, the heat transfer coefficients in the intermediate heat exchanger (IHX), and third module is applied to assessment of the natural convection in the Emergency Draining Tank (EDT). This development serves the primary estimation of the secondary and tertiary loop dimensions required by this design. They are later transferred to SIMMER-III and serve as a starting point for a more detailed simulation. In the future, these codes can be reused in the design adjustments. Additional effort to couple the existing scripts may be suggested as a future work.

The presented results show that the natural circulation in the core is sufficient to obtain the salt temperature in the heat exchanger at such level, that reasonable secondary and tertiary loops can be designed. The coolants used in the secondary and tertiary circuits (inert salt and sodium) can be modified in the future.

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Assessment of ATHLET code through its application to a LFR decay heat removal system

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Among the Generation IV reactor technologies, lead-cooled fast reactors (LFRs) have been identified as one of the suitable candidate, in terms of safety, reliability, sustainability and economics.LFRs safety is a crucial issue and the development of safety systems for protecting the plant integrity is highly demanded. For demonstrating the safety of those innovative systems, particular attention has to be payed to codes' verification and validation, because of their use in the characterization of the different reactors' components. This paper is focused on the application of the ATHLET code to a bayonet tube decay heat removal heat exchanger. The heat transfer mechanism has been investigated under different boundary conditions and by using different heat transfer correlations, as well as a parametric study of major geometrical features.The analysis has provided useful feedbacks about the ATHLET code application to LFR technologies and has allowed to identify potential improvements to the modeling of a DHR.

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Assessment of TRACE V5.0 Patch 8 using BETHSY 4.1a TC test

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The objective of the 4.1a TC (counter-part test) test carried out on the Boucle d`Etudes Thermohydraulique Systeme (BETHSY) facility was at different states to study behaviour of the primary system under single phase natural circulation, two phase natural circulation and reflux condenser mode, as well as transitions from one mode to another. The purpose of the present study was to assess the TRACE calculation of BETHSY 4.1a TC test against RELAP5 calculation and experimental data. BETHSY facility was a 3-loop replica of a 900 MWe Framatome pressurized water reactor (PWR). It was scaled down model of three-loop Framatome nuclear power plant with the thermal power 2775 MW, designed to simulate most PWR accidents of interest, to study accident management procedures and to validate the system thermal-hydraulic computer codes. The power was limited to the decay heat level, therefore the transient without reactor trip cannot be simulated. The TRACE input deck was semi-converted from the legacy RELAP5 input deck using Symbolic Nuclear Analysis Package (SNAP). The RELAP5/MOD3.2 input model of BETHSY facility from 2001 has been used for RELAP5 to TRACE conversion. For calculations the TRACE V5.0 Patch 8 and RELAP5 developmental version 3.3lj were used (the last versions

available). In 4.1a TC test the core power was 1430 kW, which is representative of 5% nominal power. The steam generators have same conditions at pressure 6.8 MPa at normal level. During the test the sequence of withdrawal of a predetermined quantity of primary coolant was repeated 14 times. Corresponding to these 14 draining phases the test covers 15 stages from 100% to 38% of reactor coolant mass inventory, where term "stage" characterizes the period between two "draining phases". Corresponding to this terminology, the sequence of events of 4.1a TC test is: stage 1.1, draining phase No. 1, stage 1.2, draining phase No. 2, stage 1.3 etc., up to the draining phase No. 14 and the final stage 1.15. The results showed that the RELAP5 and TRACE computer codes are shown to reasonably simulate single-phase natural circulation, two-phase natural circulation, and reflux condensation.

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Probabilistic margin assessment for PTS analysis during LTO

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The paper describes the probabilistic margin assessment performed during the EU APAL project. In the APAL project, an effort in the analysis of thermohydraulics uncertainties propagations to final fracture mechanics results was one of the main focus. The idea is to include in the final probabilistic analysis not only ‘mechanical’ uncertainties, described by the distributed values, like fracture toughness and artificial reference temperature, but also understand and underline the conservatism that is built in thermohydraulics analysis. As the main scenario, an ICAS transient is used, but Wilks analysis is performed to obtain a confidence bound for the transients compared to the previous study. Generated Wilks transients are used in deterministic and probabilistic margin assessment to define the maximum allowable reference temperature and year of operation. An effort is also dedicated to estimating the most probable point of failure and determining the most influential parameters. Probabilistic analysis is performed using Monte Carlo simulation, with a sampling size not less than 1e6. A comparison between probabilistic and deterministic margin and between the tangent approach and warm pre-stressing approach is made.

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Safety analyses and severe accidents and PSA

Non-ergodic ground motion model for NPP sites in Slovenia

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The seismic risk assessment and earthquake-resistant design of nuclear power plants (NPPs) can be biased because ground motion models (GMM) used in probabilistic seismic hazard analysis (PSHA) often rely on the ergodic assumption due to a lack of local strong ground motion data. To address this, we introduce a methodology for developing a non-ergodic GMM for site-specific PSHA using a local small-magnitude ground motion database. The methodology consists of three main phases: establishing the local ground motion database, modeling the non-ergodic GMM for effective amplitude spectrum (EAS) with Gaussian process regression, and converting the EAS model to the non-ergodic pseudo-spectral acceleration GMM using random vibration theory (RVT). The theoretical background and application of the non-ergodic GMM for NPP sites in Southeast Slovenia are presented. The local database consists of 1078 recordings from 130 earthquakes with magnitudes between 2.3 and 5.3. The ergodic BA19 GMM for EAS was used as the backbone to develop non-

ergodic EAS-based adjustments using Bayesian regression. The non-ergodic GMM for EAS, including adjustments for source, path, and site terms, was realized for nine frequencies, each with 108 EAS values, to capture epistemic uncertainty. The non-ergodic GMM showed reduced aleatory standard deviation and reduced hazard at the site compared to the ergodic CY14 GMM. The presented model is the first fully non-ergodic GMM considered in the PSHA for the NPP site. GEN sponsored the project.

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Improved HPC models for fuel performance predictions

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Nuclear power plants play an important role in providing clean, secure, and sustainable energy to meet the zero-emission goal. However, their operation poses unique challenges related to the durability, material performance, safety, and long-term sustainability. The design and analysis of the nuclear reactor is undoubtedly one of the most challenging due to the related great safety concerns.

The development of new and better modelling and simulation approaches can allow to better predict the fuel-cladding behaviour by considering the influence and coupling of various and different physics and scales, and the synergy between mechanical, thermal, chemical, and radioactive conditions, during steady-state and transients operation. In particular, this paper focuses on the high-performance computing (HPC) approach that can provide a more reliable prediction with complex multiscale and multiphysics modelling, in a way that is not otherwise possible by employing traditional technique.

Characteristics, parameters, advantages, and gap of reduced/surrogate model, with reference to creep and other phenomena, are presented and discussed.

Results demonstrate the advantages in obtaining faster and consistent data that can support the optimization of the plant operation.

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Application of AI methods for describing the coolability of debris beds formed in the late accident phase of nuclear reactors

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Simulations play an important role in the analysis and understanding of highly complex phenomena that occur in the late stages of a reactor accident. They can provide insight into a wide range of scenarios, such as the formation and cooling of debris beds, which can form after the failure of the lower head of the reactor pressure vessel (RPV) either inside the RPV (in-vessel bed) or in the rector cavity (ex-vessel bed). The cooling of these debris beds is paramount to the prevention or mitigation of the consequences of the reactor accident. Consequently, the research concerning debris bed formation and cooling are placed at the top tier in the priority list, which was formulated with input from prominent international experts [NEA (2018), Status Report on Ex-Vessel Steam Explosion, OECD Publishing, Paris].

The application of Artificial Intelligence (AI) has become increasingly significant in a multitude of different fields due to its ability to leverage large amounts of data to make efficient, accurate predictions, automate complex

processes, and uncover hidden patterns and insights that were previously inaccessible or overlooked. In the field of simulations, this has led to the development of predictive surrogate models that offer faster simulation times and more efficient solvers which significantly reduce the computational time and resources. This makes AI a useful tool to be used along side classical simulations to accelerate the pace of discovery and explore more complex scenarios.

Modeling the cooling of debris beds presents a significant challenge due to its complexity, requiring computationally intensive numerical simulations like computational fluid dynamics (CFD) software or specialized multi-dimensional multi-physics codes for realistic study. By employing diverse AI techniques to analyze simulated data and forecast outcomes based on initial simulation conditions, not only do we gain insights into correlations between simulation variables, but we also significantly diminish the simulation time.

In this research, the capability of AI to generalized on substantial datasets is exploited on the data collected from the COCOMO (Corium Coolability Model) code developed by the Institute of Nuclear Technology and Energy Systems (IKE) at the University of Stuttgart. COCOMO is a simulation model under development at the IKE and focuses on simulating thermal-hydraulic processes in the late phase of core degradation during severe accidents in light water reactors (LWRs). By simulating the physics of two-phase flow of steam, non-condensable gases, and water in complex geometries, COCOMO is able to determine the resulting geometries and coolability of core degradation and particulate debris formation, both in-and ex-vessel. Embedded within the German system code ATHLET-CD (Analysis of THermal-hydraulics of LEaks and Transients-Core Degradation), COCOMO replicates the impacts of potential core re-flooding and dryout scenarios in both 2D (cylindrical r-z geometry) and 3D (x, y, z) configurations.

This study conducts an examination of both supervised- and unsupervised- machine learning methodologies with the goal to advance our comprehension and predictive capabilities regarding quenching processes within packed debris beds. Supported by funding from the German Federal Ministry of Education and Research, our research aims to establish a robust surrogate model focused on forecasting quenching times. Our work encompasses the collection of a broad dataset from the COCOMO model, the identification of crucial input parameters such as corium composition, particle size, porosity, and other vital factors and the development of a fast running surrogate model on the basis of AI. First analysis of the surrogate model shows to have reasonable agreement with the simulation results.

The insights gained from the developed surrogate model have the potential to greatly enhance the efficiency and accuracy of simulations, facilitating informed decision-making and risk mitigation strategies during severe accident scenarios.

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Safety analyses and severe accidents and PSA

Evolution of Safety Concepts in Pressurized Water Reactors

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Nuclear power plant safety remains a crucial concern in present society, prompting continuous advancements in reactor design and safety systems. This study meticulously examines and compares safety systems across three generations of pressurized water reactors: Generations II, III, and III+.

The Generation II KWU reactors laid the foundation for today’s reactor safety with the successful integration of the "Defense in Depth" approach while considering redundancy and separation of vital components. The featured systems present a balanced net of safety to counteract design basis accidents. However, flaws become apparent regarding their reliance on active components during combined stress scenarios and in mitigating core melt scenarios.

Advancing to Generation III and III+, major improvements were achieved in critical safety areas with the APR-1400 and the EPR. These reactors offer simplified safety systems, reinforced containment structures, and innovative measures targeting severe accident scenarios, notably core melt accidents. The APR-1400's adoption

of in-vessel retention for molten corium mitigation represents a generational leap forward, while the EPR's core catcher provides a sophisticated system of ex-vessel cooling and corium retention for reactors with a high thermal rating.

With the Generation III+ AP1000, a step further was taken with its radical design philosophy. By harnessing natural forces such as gravity and convection, this reactor essentially eliminates the need for any major active safety components and intervention by plant personnel, showcasing the future of passive reactor safety.

Through a comparative analysis, this study unveils major achievements and quantum leaps in pressurized water reactor safety. The evolution from Generation II to III and III+ reactors outlines a continuous path of safety improvements and standards, laying the groundwork for a future defined by unmatched safety in nuclear power generation. This comprehensive examination of safety systems and advancements offers valuable insights into the trajectory of nuclear safety and its implications for future reactor designs.

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Safety analyses and severe accidents and PSA

Uncertainty analysis of simultaneous SBO and LBLOCA scenario considering chemical reactions

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An uncertainty and sensitivity analysis of a severe accident scenario in the Krško NPP was performed, applying the MELCOR code version 2.2 for the accident simulation and the SUSA code for the sensitivity analysis. As the initiating event, a strong earthquake was considered, resulting in a simultaneous station black-out (SBO) and large break loss-of-coolant accident (LBLOCA). It was assumed that only passive safety systems are available. The uncertain input parameters were selected based on the EC MUSA (Management and Uncertainties of Severe Accidents) project and the State-of-the-Art Reactor Consequence Analysis performed recently by the United States Nuclear Regulatory Commission and Sandia National Laboratories. The following subset of uncertain parameters was considered: heat transfer coefficient from debris to lower head, aerosol dynamic shape factor, debris to cavity heat transfer at the bottom surface of the debris, fraction of iodine in gaseous form, fraction of caesium which reacts with molybdenum.

As in MELCOR the chemical reactions cannot be properly modelled directly, they were considered indirectly by adjusting the inventory of the Cs, I2, CsI, Mo, and CsM radionuclides classes in the core to already reflect the consequences of chemical reactions. Three cases were considered: (i) default core inventory without considering chemical reactions, (ii) consideration of Cs + I reaction, and (iii) consideration of Cs + I and Cs + Mo reactions. For each case 120 calculations were performed, sampling the uncertain model parameters according to the assumed probabilistic distributions. More than 93 calculations for each case were successful, thus fulfilling the criterion for the 95 % probability content with a 95 % confidence level according to the Wilks formula. For the analysed simulation results the Spearman's rank correlation coefficients for the considered uncertain model parameters were calculated. In the paper the results of the performed analysis will be presented and discussed. The focus will be on the behaviour of radionuclides. Based on the study some suggestions for the optimal radionuclide's modelling approach will be provided.

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Comparative Analytical Study of High Temperature Oxidation of ATF FeCrAl, Cr-Coated Zr-Based Alloy, Crome-Nickel Alloy and SiC-Based Composite Claddings in Steam Atmosphere

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Currently, the use of perspective advanced tolerant fuel (ATF) claddings is considered as one of encouraging ways to strengthen the reliability, safety and performance of nuclear fission generation.

The most promising perspective ATF cladding candidates for possible application in commercial nuclear power plants (NPPs) throughout the world include:

- *FeCrAl* alloy cladding;
- zirconium-based cladding with protective chromium coating (*Zr/Cr* cladding);
- chrome-nickel alloy cladding with high *Cr* content;
- SiC-based ceramics composite cladding.

The *FeCrAl* alloy, the chromium, the chrome-nickel alloy and SiC-based composite have excellent characteristics of corrosion and oxidation resistance compared to zirconium both for the NPP normal operation temperatures and high-temperature conditions. It is very important for the nuclear safety including the resistance to design-basis and beyond-design-basis accidents at NPPs.

However, the worsening of *FeCrAl* cladding oxidation characteristics is reported when approaching the melting temperature of *FeO* (T=1371°C). Also, recent experimental data showed that in the temperature range close to upper limit of design-basis accident (T=1200°C) and higher there is a considerable worsening of *Zr/Cr* cladding protective properties. The chrome-nickel ATF-cladding is expected to loose its protective properties when the temperature approaches its liquidus temperature T=1370¸1400°C. The barrier effect of SiC/ SiC cladding (the only non-metallic candidate among ATF-cladding considered) is destroyed at temperatures higher than 1700°C.

In this paper, the new advanced models of high-temperature oxidation of *FeCrAl*, *Zr/Cr*, *Cr-Ni* and *SiC*/*SiC* cladding are developed. Special attention is paid to critical phenomena taking place in those claddings during high-temperature oxidation. For example, the formation of *FeO* melt leads to acceleration of oxidation and hydrogen generation when approaching to temperature T=1371°C and above in *FeCrAl* cladding. The *Zr/Cr* cladding oxidation model is based on simultaneous solution of oxygen and zirconium diffusion equations in different layers of the cladding. The role of *Cr-Zr* interdiffusion with subsequent influence on degradation of protective properties is considered. The expected degradation of protective properties for *Cr-Ni* cladding is connected with enhancement of oxygen diffusion coefficient in Cr_2O_3+NiO oxide for temperatures close to liquidus temperature of chrome-nickel cladding. The evaporation of silicon oxide at high temperatures is one of important factors during high-temperature oxidation of *SiC*/*SiC* cladding.

The comparative analytical and numerical calculations of these ATF-claddings behavior under high-temperature oxidation are performed. The comparison of calculated results with available experimental data is conducted. The reasonable agreement between calculated and experimental data is observed.

Despite the existence of obvious mechanisms leading to loosing of protective properties at high temperatures, one can make a conclusion that the application of *FeCrAl*, the chromium-coated Zr-based cladding, the chrome-nickel claddings and SiC-based composite cladding may be optimistic for considerable upgrade of safety level for NPPs especially for design-basis-accident conditions.

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An Overview of Seismic Design Parameters for Design of Nuclear Power Plants

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The paper will present an overview of various regulatory documents, guidelines, standards, and codes pertaining to the seismic design of Nuclear Power Plants (NPPs). The overview will focus on IAEA, WENRA, EUR, EPRI, US NRC and Slovenian regulatory documents. The primary focus will be on the requirements related to seismic design levels, particularly emphasizing the definitions of mean annual frequencies (MAFEs) and the corresponding return periods (TR) of seismic hazard events. Some documents specify exact values for MAFEs, while others define them within a range. Additionally, certain documents provide minimum seismic design levels, typically in terms of minimum peak ground acceleration (PGA). The precise delineation of the seismic levels is crucial for assessing uniform hazard spectra (UHS) and creating design response spectra (DRS), which serve as the design basis for NPPs. It is imperative that the NPP's structures, systems, and components (SSCs) are designed, built, and operated to withstand both the Design Basis External Hazards (DBEH) and infrequent, severe external hazards, while maintaining an adequate safety margin.

The paper will also outline the performance-based approach, which seeks to estimate the likelihood of achieving either acceptable or unacceptable performance during a specific hazard event, such as the likelihood of encountering unacceptable performance due to site-specific vibratory ground motion. Consequently, this approach aims to meet a predefined damage level as its final frequency objective or target. The approach offers a systematic method for evaluating the performance capacity of a building, system, or component. An integral part of this approach is the Probabilistic Seismic Hazard Analysis (PSHA), which primarily yields the estimation of the Uniform Hazard Spectra (UHS).

The final goal of the paper will be to assess various seismic design levels by examining regulatory documents, guidelines, and standards. These seismic design levels are primarily determined by the values of mean annual frequencies (MAFEs) in most documents, and some also specify minimum requirements for Peak Ground Accelerations (PGA). According to established guidelines and standards, seismic design levels can be categorized into three groups:

1. The first group corresponds to Operational Basis Earthquakes (SL-1 or OBE) and Low-Level Earthquakes (LLE) and is defined for return periods (TR) ranging from 100 to 1000 years.
2. The second group pertains to Safety Shutdown Earthquakes (SL-2 and SSE) and Design Basis hazard Events (DBE). These seismic levels are defined for return periods (TR) between 1000 and 10 000 years, except for SL-2 according to IAEA, which may extend up to 10 000 or even 100 000 years.
3. The third group is associated with Rare and Severe Hazard Events (RSEH) and Beyond Design Basis Events (BDBE). These seismic levels are defined for return periods (DBE) exceeding 10 000 years but remaining below 100 000 years. Some guidelines base the RSEH and BDBE levels on SL-2 or SSE seismic criteria while considering additional factors or safety margins.

For a more comprehensive comparison, the assessment of seismic levels is outlined by considering specified return periods sourced from reviewed documents along with associated PGAs. The PGA levels are estimated considering the valid PSHA study from 2004 which was performed for NEK (Nuklearna elektrarna Krško).

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Simulation of vapour explosions in combined melt-jet-breakup and stratified configuration

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A vapour explosion is a possible threatening consequence of a fuel-coolant interaction. This phenomenon can occur during a severe accident in a nuclear power plant, when the molten reactor core may come in contact with the coolant.

Based on previous research, an important uncertainty regarding vapour explosion assessment was raised. Past research was devoted to either melt-jet coolant pool configuration or stratified configuration. However, an intertwined configuration can be a realistic condition. Combined stratified and melt-jet-breakup configuration is studied within this research.

The first objective of the research is to implement the developed modelling approach for the premixing phase in combined stratified and melt jet configuration into the computational fluid dynamic code for severe accidents MC3D (IRSN, France). Mixing of melt and coolant prior to vapour explosions largely defines the amount of melt, participating in the vapour explosions. Different processes for melt-coolant mixing and other fuel-coolant interactions are being combined, which enables us to simulate complex phenomena and improve the understanding of fuel-coolant interaction in realistic conditions.

Secondly, the patch of MC3D code being under development is used to validate the approach on the available experimental results. A recent experiment PULiMS E6 (KTH, Sweden) is being simulated, showing the importance of adequately describing the premixing phase by combining the melt-jet-breakup and stratified configuration melt-coolant mixing.

Only simulations with the validated code will enable the estimation of the premixing phase in combination of melt-jet-breakup and premixed layer formation during melt spreading as well as potential consequent vapour explosion, which is of high importance in nuclear safety.

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Identification of PTS scenarios for two-loop PWR

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During nuclear power plant operation, the wall of reactor pressure vessels (RPVs) is exposed to neutron radiation. This leads to localized embrittlement of the vessel steel and weld materials in the core area. If an embrittled RPV contains a critical-sized flaw and experiences severe system transients, the flaw could propagate rapidly through the vessel, leading to a through-wall crack and challenging the RPV's integrity. The severe transients of concern are known as pressurized thermal shock (PTS). They are characterized by a rapid cooling of the internal RPV surface in combination with depressurization of the RPV. The purpose of this study is to identify overcooling scenarios for selected two-loop pressurized water reactor (PWR), which potentially pose a challenge for PTS. The study is part of the Slovenian Research and Innovation Agency project L-4432, which aim is to demonstrate a methodology for advanced safety analysis for the most limiting overcooling event. Namely, the most limiting overcooling scenario will be selected based on the simulations of scenarios, which are to be identified in this study.

In the paper some PTS background will be presented first. Then the PTS scenarios selection method will be presented. The candidate groups of scenarios will be determined based on the review of public literature on PTS studies. The main criteria for the choice of scenario groups are initial cooling rate, minimum temperature of transient and pressure retained in the primary system. For the identification of scenarios specific to two-loop PWR, an analysis of candidate groups of scenarios will be done, also considering the published thermal-hydraulic safety analyses considering specific design (e.g. spectrum of cold leg loss of coolant accidents) and possible influence of operator actions.

The result of this study will be the list of severe overcooling scenarios (including their description and credited operator actions) for later thermal hydraulic safety analyses of two-loop PWR using TRACE computer code.

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TRACE simulation of hot leg LOCA spectrum in two-loop PWR

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The severe transients of concern known as pressurized thermal shock (PTS) are characterized by a rapid cooling of the internal RPV surface in combination with depressurization of the RPV. PTS can occur during several overcooling scenarios, including loss of coolant accidents. The purpose of this study was to perform the calculation of hot leg loss of coolant accident (LOCA) spectrum calculations using advanced TRAC/RELAP Advanced Computational Engine (TRACE) computer code. The reactor selected was two-loop pressurized water reactor (PWR). The TRACE input deck developed in 2023 was used for calculations. All important systems and components were modelled, including reactor coolant system, secondary side, control systems, reactor protection system logic and safety systems. The break occurred in the hot legs at full reactor power. One train of active emergency core cooling systems was available (one high pressure safety injection (HPSI) pump, one low pressure safety injection (LPSI) pump) and both accumulators. Both motor driven (MD) auxiliary feedwater (AFW) pumps were assumed available. After break occurrence the reactor trips on (compensated) low pressurizer pressure signal (12.99 MPa), which further causes the turbine trip. The safety injection (SI) signal is generated on the low-low pressurizer pressure signal at 12.27 MPa. On SI signal active safety systems like HPSI pump, LPSI pump and both MD AFW pumps start. When primary pressure drops below 4.96 MPa, both accumulators start to inject. When primary pressure drops below 1.13 MPa, LPSI pump start to inject.

To assess the results, code to code comparison has been done, i.e., the results obtained by TRACE computer code were compared to results obtained by RELAP5 computer code.

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Systematisation of knowledge on severe accident phenomena and experiments for preservation and transmission

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The past decades have witnessed intense research in severe accidents in light water reactors concerning all accident phenomena (in-vessel phenomena, ex-vessel phenomena, containment phenomena, source term behaviour). In addition to already existing knowledge, this research has produced an additional vast amount of theoretical and experimental results. However, despite the capitalization of knowledge conducted through codes development and validation, further attention should be given to preserve and pass on the expertise and capabilities for the research to be addressed in the coming decade to new generations in the severe accident field. That is, a researcher new in any topic within severe accidents is faced (among others) with the following questions:

- what is the state-of-the-art in the knowledge about specific phenomena ?

- which experiments have been performed within that topic ?

Within the European SEAKNOT project (2022-2026), among other tasks, suitable documentation to fulfil those needs is being composed. A Phenomena Identification and Ranking Table (PIRT) on phenomena during a severe accident is being prepared. In addition to summarize the phenomena as done in various states of the art, the PIRT will also provide an assessment of the knowledge about the listed phenomena and of their safety significance. A directory of experimental data related to severe accidents is also being prepared that, in addition to provide a comprehensive list, will also include relevant information for the use of specific data in further research. The methodologies to prepare the PIRT and the experiment database directory will be described in the proposed paper.

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Analysis of Current Flood Protection of the Expected JEK2 Site

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While designing nuclear power plants, it is crucial to consider both external and internal hazards that may pose significant risks to safe operation. One of the important external hazards to consider is flooding.

The central area of the second nuclear power plant unit, referred to as JEK2, is already protected against a Probable Maximum Flood (PMF) of river Sava, with a flow rate of 7.081 m^3/s, by existing flood embankments. Analyses have determined that safety against Sava's flow rates of up to 11.130 m^3/s is ensured without additional upgrades.

Besides Sava, the flood safety of the JEK2 central area is also impacted by stream Potočnica. Currently, the JEK2 area is not fully protected against a PMF of stream Potočnica with a flow rate of 123 m^3/s. This will be ensured after the safety measures planned by the Krško Nuclear Power Plant (NEK) for 2024 are completed.

Previous hydraulic analyses did not consider structures in the JEK2 area. Therefore, new comprehensive hydraulic analyses are required to confirm the flood safety of the JEK2 area or to propose specific additional measures to ensure or improve it.

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Evaluation of meteorological data for the site of the new nuclear power plant Krško 2 (JEK2)

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The aim of meteorological data evaluation was to determine the necessary meteorological data essential for preparing the Site Safety Analysis Report, the Preliminary Safety Report, and other documentation for the new Krško Nuclear Power Plant (JEK2). The analysis of meteorological data is crucial for designing the heat sink systems of JEK2, where the ultimate heat sink is air. Key meteorological data included dry bulb temperature, relative humidity, and wet bulb temperature, which served as the design value for the safety cooling system and the system for discharging waste heat from the secondary system during normal operation.

The project involved statistically processing data on air temperature, relative humidity, and wet-bulb temperature to assess climatic conditions. Data were sourced from the Slovenian Environment Agency's national meteorological network (ARSO) and the Ecological Information System of the Krško Nuclear Power Plant (EIS NEK), with key inputs from automated meteorological stations (AMP) adhering to World Meteorological Organization standards.

Half-hourly average data were used to calculate wet-bulb temperatures, forming the basis for further statistical analyses. Analysis of extreme values employed the Generalized Extreme Value (GEV) distribution to determine values for different return periods. Future climate change projections up to 2100 were considered, based on ARSO scenarios, indicating potential increases in average annual temperatures and extremes, influenced by greenhouse gas emissions and MEIS research series of weather forecasts.

The results are crucial for understanding localized and broader climatic changes, thereby informing the planning and development of JEK2, ensuring the representativeness and long-term stability of the measurement locations, with AMP Stolp identified as the most suitable.

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Analysis of atmosphere composition at severe accident conditions in a pressurized water reactor containment with lumped-parameter description

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The issue of atmosphere flammability during a severe accident due to combustible gas generation has been considered for a long time, especially after the hydrogen explosion at the Three Mile Island (USA) nuclear power plant (NPP) in 1979 and again later after the explosions at the Fukushima (Japan) NPP in 2011. Although the basic mechanisms that lead to hydrogen and carbon monoxide generation and to formation of a flammable mixture are well understood, the prediction and analyses to devise suitable mitigation measures still present difficulties for an actual NPP, due to:

- demanding modelling of detailed phenomena in reactor core degradation and later in molten core - concrete interaction;

- large volumes of NPP containments which makes it impractical to simulate phenomena on the local instantaneous scale or to perform numerous simulations using three-dimensional codes with subgrid modelling.

For these reasons, system severe accident codes that use (relatively) simplified modelling of reactor core degradation and lumped-parameter description of NPP containment are still being used despite their limitations.

Within the Euratom AMHYCO project (Grant Agreement n°945057), a series of simulations of the evolution of the NPP containment atmosphere at accident conditions has been performed. The ultimate goal of the simulations is to generate a database for assessing the existing Severe Accident Management Guidelines with regard to combustion risk.

The considered containment is a generic model of a so-called "Western" type pressurized water reactor (PWR), developed at the Polytechnic University of Madrid (Spain). The sources of flammable and other gases were obtained from simulations of a large-break loss-of-coolant accident performed by CIEMAT with the MELCOR code and simulations of a station black-out accident performed by IRSN with the ASTEC code.

In the proposed paper, results of simulations of the evolution of the containment atmosphere composition using the lumped-parameter description within the ASTEC code by JSI are presented. Parametric simulations were also performed with the action of mitigation systems (Passive Autocatalytic Recombiners and containment sprays) to analyse their influence in preventing the atmosphere composition to evolve beyond flammability limits.

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Influence of Model Parameters on Pin Power Distribution Due to Fuel Assembly Bowing

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One of the main approaches to ensuring nuclear safety in neutron calculations is the consideration of engineering factors. These factors define the margin of individual parameters in the power distribution within the core up to their safety limit, which must be respected in the fuel assembly and core design and monitored during the operation of the fuel cycle. The resulting engineering factor consists of many components; one component of this set is investigated in this report, specifically the evaluation of fuel assembly bowing or displacement and the subsequent impact power distribution on the edge rows of fuel pins. This work focuses primarily on the effect of the model's external parameters on the conservativeness of the solution. Results from this study can be used as input parameters for special detailed analysis.

New reactor designs and small modular reactors

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New reactor designs and small modular reactors

Fuel for the Future – Two Reactor Concepts to Tackle the Burden of Accumulating Spent Fuel and High-level Radioactive Waste

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The reprocessing and recycling of spent nuclear fuel (SNF) is the basis of a sustainable energy strategy that aims to "close" the nuclear fuel cycle and thus reduce the demand for resources, while at the same time optimizing the management of the radioactive waste produced, i.e. reducing the quantity and long-term radiotoxicity of the final waste. Promising innovative concepts for the utilization of SNF include so-called "waste to power" reactors with a fast spectrum, in which the majority of the fissile nuclides (uranium, plutonium) are also used for the generation of electricity and only the fission products, which have significantly shorter half-lives compared to uranium and all long-lived actinides, have to be deposited in a deep geological repository after separation.

We will present two efficient transmutation scenarios: one based on a Fast-Spectrum Molten Salt Reactor (FS-MSR) with chloride salt, and the other based on an Accelerator Driven System (ADS) with a subcritical lead-cooled reactor. The conceptual differences and R&D challenges to move from "paper reactors" (concept studies) to prototypes are highlighted. Furthermore, for a given SNF inventory of a LWR fleet, a comparative analysis of the two reactor concepts is performed, focusing on the radiotoxicity and thermal power of the resulting final waste.

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New reactor designs and small modular reactors

Heat-Only Small Modular Reactors Vs. Nuclear Combined Heat and Electricity: Concepts and Economics

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There is a growing interest in the heat-only small modular reactor concept for clean, efficient, and economic district heating applications. On the other side, there is an argument that upgrading the existing nuclear power plants to a combined heat and electricity generation (CHP) mode could be a competitive option. The upgrade of NPPs to CHP mode requires several pieces of equipment, such as heat exchangers and often a long-distance heat transportation pipeline. Furthermore, the maximum electrical power generation capacity is decreased when the power plant is operated in CHP mode. While the capital cost of heat-only SMR could be higher, the optimum siting could reduce the heat transportation construction and operation cost, and the integration with secondary thermally driven applications could significantly improve the overall economics of the system. In this context, this paper investigates some potential EU cities to be supplied with clean district heating, discusses the advantages, disadvantages, and limitations, and provides a systematic method for estimating the economics of the heat-only Small modular Reactors vs. combined heat and electricity generation scenarios.

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Development of small modular reactor cooled by supercritical water within the ECC-SMART project

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The ECC-SMART: Joint European-Canadian-Chinese development of small modular reactor technology is Euratom Horizon 2020 project. This project started in September 2020 and it will end in February 2025. The ECC-SMART project reunites specialists from 17 countries and three continents. It profits from the cooperation and knowledge obtained within previous international projects.

The ECC-SMART aims to make a real progress towards the design of advanced small modular reactor cooled by supercritical water (SCW-SMR). Use of passive safety systems is one of the most important principles on which this type of reactor will be designed. This contribution describes the implementation of the project and how the stated objectives have been fulfilled within four technical work packages (Material testing, Thermal-hydraulics and safety, Neutronics and reactor physics, and Guideline synthesis and pre-licensing studies). The most interesting results obtained so far are presented as well. However, the successful project completion is indicated by submitting 30 deliverables and 20 milestones. In addition, a high number of students were successfully involved and supported the project´s results which were also reflected in published numerous peer-reviewed open-access scientific papers. Overall ECC-SMART project can be considered as an important contributor to the new type of advanced SMRs since SCW-SMR could be utilized in other industrial processes and technologies.

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3D Reactor Core Simulation of ACP100 SMR

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Base on the reactor core of ACP100 small modular reactor nuclear power plant, a 3D neutronics calculation model based on fuel assembly cross section calculation and fitting is established with Apros simulation software. First the 2D homogenization of cross-section data was calculated in Monte Carlo code SCIENCE V2 developed by AREVA. The cross-section data of each type of fuel assembly is calculated according to their enrichment, burnable poisons and burnup. And the reactivity feedback coefficient is fitted with changes in cross-section data under different boron concentration, reactor coolant density and fuel temperature. The boundary of reactor is represented by albedo element in Apros. The steady state simulation at full power is simulate in Apros and compared with SCIENCE code. Relatively good accordance is acchieved between Apros and SCIENCE code. Further dynamic transit analyses are performed: including full power reactor trip, symmetrical control rod bank insertion and asymmetrical insertion of one control rod.

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Review of Existing and Emerging Nuclear New Build Business Models

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This paper will review existing and emerging nuclear newbuild business models and in particular:

1. The buyer-led structures featuring ownership and offtake stakes (Finland, France)
2. Hybrid Structure where a portion of the output is sold on a contracted basis and the remaining capacity is sold into the market / grid (Turkey)
3. Government-backed owner-operators with market buyers (UK HPC/CfD; UK RAB-SizewellC, Czech Republic, UAE)
4. Government to government (Poland and others)

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Preliminary Design and Optimization of the Heat Pipe Heat Exchanger (HPHX) for a 5 MWth Heat Pipe-Cooled Micro Modular Reactor (MMR)

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The growing global demand for energy underscores the need to explore innovative and environmentally sustainable solutions. One notable development in this field is the emergence of Micro Modular Reactors (MMRs), a subset of Small Modular Reactors (SMRs), specifically designed to produce less than 10 MW of electrical output. Among these, the Heat Pipe-Cooled MMR has attracted significant attention. With a thermal power of 5 MW, the analysed design of Heat Pipe-Cooled MMR utilizes a stainless-steel monolithic core, divided into six blocks, which is drilled to accommodate uranium dioxide fuel pins and liquid potassium heat pipes in a honeycomb configuration. These 1224 heat pipes play a crucial role in dissipating heat from the core by vaporizing the liquid within them, with heat being released in the condenser section. The power conversion system, currently envisioned with air or supercritical CO_2 as working fluid, aims to efficiently convert the extracted heat into power. The Heat Pipe Heat Exchanger (HPHX) is a fundamental component in this process.

This study presents a preliminary investigation of the HPHX design. This is envisioned to be an annular flow heat exchanger, where each heat pipe is individually cooled with a dedicated annulus. A MATLAB code was developed to calculate the annulus diameter that allows achieving the required turbine inlet temperature and ensures the same pressure drop in the different annuli, considering the geometrical constraints and the thermodynamic boundary conditions. The heat pipes, modelled as constant temperature boundary conditions, experience different power and have different wall temperatures depending on their location in the reactor. For this reason, it is important to balance the pressure drop across all the channels, to avoid flow instabilities and ensure structural integrity. The code allows to calculate the mass flow rate, the outlet pressure, the outlet temperature and other thermodynamic parameters for each type of annulus.

The main requirement for the HPHX is to effectively remove heat from the heat pipes. Other important factors to consider include the pressure drop across the heat exchanger and its dimensions. In the first case, a significant increase in the pressure drop results in higher compressor work, which decreases the cycle efficiency. In the second case, since the MMR should have a compact design, the HPHX should not be too large. To consider these factors, an optimization analysis is conducted to find the optimal annulus diameter. The performance of air and sCO_2 are compared, highlighting the superior heat transfer performance of sCO_2, which requires a smaller heat exchanger for the same relative pressure drop.

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Sustainable hydrogen production from nuclear energy

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The rapid increase in global warming, with a global temperature increment of 2.7 °C at the end of 21st century, makes urgent a fast transformation in energy infrastructure and transportation sector to mitigate greenhouse gas emissions. The hydrogen is at present considered one of the promising resources that could allow to meet the global increase in energy demand by matching zero-carbon goal.

Despite several technologies have been proposed over the past years for hydrogen production, the nuclear energy seems the most promising for application beyond electricity generation. It can be key enablers to produce hydrogen thermochemically in a clean and efficient manner. Especially the nuclear advanced reactors, that operate at very high temperatures (VHTR) and are characterised by coolant outlet temperatures ranging between 550-1000°C, seem the most appropriate for a coupling with hydrogen production processes.

This paper describes the potential use of nuclear and renewable generation in coordinated and coupled configurations to support the hydrogen production. Operating conditions, energy requirements and thermodynamic performance are described. Moreover, gaps that require additional technology and regulatory developments are outlined. The intermediate heat exchanger that is the key component for the integration of nuclear-renewal hybrid energy systems, is studied varying the thermal power in order to determine physical parameters that are for feasibility study. This latter was carried out by using the IAEA HEEP code. Preliminary results are presented and discussed.

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Teplator in Sea Water Desalination – Technical and Economical Evaluation

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Teplator is an innovative concept of SMR (Small Modular Reactor). This concept deploys already used nuclear fuel for cheap heat production for district heating purposes. Hence the parameters of produced heat are quite low (130 °C) and its use in for industrial purposes is quite limited, there are few possibilities where cheap and low parameters heat can be utilized. Nuclear powered sea water desalination is one of such possibilities. In this paper we assume technical aspects and challenges for Teplator and desalination plant coupling. Technical evaluation and economic study using the Desalination Economic Evaluation Program (DEEP) made by the IAEA.

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Emerging Issues in Ultrasound Safety Evaluations for Liquid Metal-Cooled Small Modular Reactors (SMRs)

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Safety inspections of Small Modular Reactors (SMRs) utilizing liquid metals like lead and sodium as coolants pose unique technical and engineering challenges distinct from those faced by traditional water-cooled nuclear reactors. This paper explores the complexities and barriers associated with ultrasound safety inspection processes specifically designed for SMRs employing these advanced coolant technologies. Drawing on comprehensive results and innovations from the INETEC collaborative project, which centers on a novel liquid metal-cooled SMR concept and prototype, we discuss the physical and chemical characteristics of lead and sodium that impact inspection procedures.

The key aspects of these coolants include high opacity, corrosivity, the need for high operational temperatures, and the presence of ionizing radiation. These factors significantly influence the development and application of ultrasound non-destructive examination (NDE) techniques. The high opacity of lead and sodium hinders the penetration of ultrasound waves, making it challenging to obtain clear images for inspection purposes. Additionally, the corrosive nature of these metals can degrade the materials and components of ultrasound inspection tools, reducing their lifespan and effectiveness.

High operational temperatures further complicate ultrasound inspections. Standard ultrasound equipment often cannot withstand the extreme heat, necessitating the development of specialized sensors and materials capable of operating reliably under these conditions. Moreover, the presence of ionizing radiation in SMR environments can interfere with electronic components of ultrasound equipment, leading to potential malfunctions or inaccurate readings.

This paper initially assesses current ultrasound NDE techniques tailored for these demanding environments, highlighting significant limitations and potential risks. One major limitation is the accessibility of critical reactor components for inspection. The compact design of SMRs, coupled with the use of liquid metal coolants, restricts the placement and movement of ultrasound sensors, making it difficult to thoroughly inspect all necessary areas. Sensor performance is another critical concern. The extreme conditions within liquid metal-cooled SMRs can impair sensor accuracy and reliability, raising concerns about the effectiveness of current NDE methods in detecting potential issues.

To address these challenges, we propose a preliminary strategic framework aimed at overcoming these obstacles through targeted technological advancements, enhanced regulatory measures, and improved training protocols. Technological advancements are crucial for developing more resilient ultrasound inspection tools. This includes creating and customizing sensors capable of withstanding high temperatures and corrosive environments, as well as improving the penetration capabilities of ultrasound waves in opaque liquids.

Enhanced regulatory measures are also essential to ensure the safety and reliability of SMRs. These measures should focus on establishing stringent standards for the design and maintenance of ultrasound inspection equipment, as well as setting clear guidelines for regular inspections and safety assessments. Additionally, regulatory bodies should encourage the continuous development and adoption of advanced NDE techniques to keep pace with technological advancements.

Improved training protocols are vital for preparing inspection personnel to operate effectively in these challenging environments. This involves providing comprehensive training on the unique properties of liquid metal coolants and the specific requirements of ultrasound inspections in SMRs. Training should also cover the proper handling and maintenance of specialized equipment to ensure long-term reliability and accuracy.

This paper offers a foundational roadmap and strategic insights for advancing safety inspection capabilities in this promising new category of nuclear reactors. By addressing the unique challenges posed by liquid metal coolants, we aim to enhance the overall safety and operational efficiency of SMRs. This, in turn, will contribute to the broader adoption of SMRs as a viable and sustainable option for meeting future energy needs.

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Study of optimization methods for performance design of nuclear district heating systems

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The incorporation of nuclear power into district heating systems introduces novel challenges and considerations for system design. The unique operation characteristics of small reactors, which serve as suitable heat sources for such systems, necessitate a specialized design approach for the entire district heating infrastructure. Achieving the most cost-effective outcome involves strategically siting district heating systems with nuclear heat sources alongside other power components. These components are instrumental in optimizing the utilization of heat generated by the reactor while accounting for variations in heat demand throughout the year.

This optimization involves the integration of thermal energy storage systems to accommodate excess heat and technologies for converting surplus heat into power, such as Organic Rankine Cycle (ORC) or Kalina cycle systems. Additionally, auxiliary and backup heat sources are essential for addressing situations of low thermal energy demand or primary heat source shutdown.

A fundamental challenge in system design lies in accurately estimating the performance requirements of all these components: the nuclear reactor, auxiliary heat sources, conversion technologies, and thermal energy storage. Utilizing appropriate optimization methods is crucial for this task.

This paper presents two optimization methodologies for enhancing the performance design of nuclear district heating systems: Mixed Integer Programming (MIP) optimization and Genetic Algorithm (GE) optimization. Both methods are applied to optimize the same nuclear district heating system described previously, utilizing real one-year heat consumption data. The paper discusses and compares the optimal model settings, initial conditions, and input parameters for both approaches. Furthermore, it presents and critically evaluates the results obtained from each method, highlighting their respective advantages and disadvantages.

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Analysis of different types of electrical power substations for JEK2

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The demand for electricity in Slovenia, as well as globally, has been increasing over the years. This is evidenced by an increase in industrial consumption and a trend towards higher usage of both electrical and thermal energy, subsequently escalating the need for new sources of electricity. When selecting an energy source, low greenhouse gas (GHG) emissions are a priority, and it must not have adverse effects on the environment or human health. Considering these criteria and the need for an energy source that could stabilize Slovenia's electrical power grid for an extended period, the most suitable option is the construction of a new nuclear power plant (NPP), JEK2. From a long-term perspective, in terms of electricity demand, environmental impact, and GHG emissions, nuclear technology stands out as one of the cleanest energy sources. Near Krško, in Vrbina, alongside the existing NPP, the construction of a new, more advanced unit is considered. The connection of this new unit necessitates a comprehensive upgrade of the switchyard.

Prior to constructing the new NPP in Krško, it is essential to ensure the adequate evacuation of the electricity generated by JEK2. Currently, the existing 400 kV transmission lines are underutilized, providing sufficient capacity for energy evacuation, particularly towards Ljubljana (Krško-Beričevo). A more significant challenge is the establishment of a suitable switchyard, constrained by both spatial limitations and the initial investment

required. Presently, two options are being considered. The existing 400/110 kV air-insulated switchyard (AIS) in Vrbina will require expansion and modification to integrate the new JEK2 unit. Alternatively, it may be necessary to construct a new 400 kV and 110 kV switchyard, employing a modern gas-insulated switchyard (GIS) configuration.

However, GIS technology brings an environmental consideration, as it uses SF_6 gas as an insulator and interrupting medium. SF_6 is a potent greenhouse gas that is already prohibited in most applications, with probable anticipated limitations or bans on its use in switchgear by 2031. Significant progress has been made by various manufacturers in developing alternative media to SF_6 for use in electrical switchgear and transmission lines. However, these products are at varying technological readiness levels (TRL). Solutions using SF_6 alternatives already exist and are commercially available at low and medium voltage levels. Based on ongoing development, research, and legislative trends, advancements in the application of these alternatives at medium and high voltage levels are anticipated in the coming years.

In the full paper, following a brief introduction, an overview of AIS and GIS technologies will be provided. The subsequent chapter will discuss SF_6 gas, including its advantages, disadvantages, alternatives, and legislation. The final chapter will present following two technical solutions for integrating JEK2 into Slovenia's electrical power grid.

Technical solution A: This solution involves integrating JEK2 at the 400 kV voltage level into the existing AIS switchyard and at the 110 kV voltage level into a new 110 kV GIS switchyard, which is planned to be constructed west of the current 110 kV AIS switchyard. Consequently, this option includes modifications to the configuration of the 400 kV and 110 kV transmission lines connected to electrical substation in Vrbina, Krško. This variant is feasible only for the construction of a JEK2 unit with a capacity of up to 1200 MVA. If a larger unit (1400 MVA or 1650 MVA) is to be constructed, the technological infrastructure would occupy the space of the existing 400/110 kV switchyard, necessitating the construction of a new 400/110 kV switchyard.

Technical Solution B: This solution entails constructing new 400 kV and 110 kV switchyards, both utilizing GIS technology, at a new location east of the access road to NEK and north of the existing parking areas adjacent to NEK. The proposed site for the new 400/110 kV switchyard and associated transformers is of a suitable size to accommodate construction at multiple micro-locations and orientations, depending on the configuration of the 400 kV and 110 kV transmission lines.

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GEN energija in pursuit of Small modular reactor's

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To establish its position as the leading promoter of new nuclear technologies in Slovenia, GEN energija has initiated several strategic tasks focused on Small Modular Reactor (SMR) technologies. The goal is to monitor new technologies, prepare feasibility studies on SMR technologies, with emphasis on the most developed technologies, determine their suitability for use in Slovenia and identify potential locations for their use. To this date several internal technical reports have been made on this topic. GEN energija is also a partner in the project Phoenix which will be implemented within the framework of the U.S. Department of State's Foundational Infrastructure for the Responsible Use of Small Modular Reactor Technology (FIRST) Program in cooperation with the U.S. Department of Commerce's Small Modular Reactor Public-Private Program (SMR PPP), which aims to promote transatlantic cooperation to deploy SMRs in Europe and Eurasia. Gen energija together with Slovenian stakeholders (Ministry of the Environment, Climate and Energy, HSE Group, Šoštanj Thermal Power Plant, Slovenian Nuclear Safety Administration, Slovenia's combined transmission and distribution system operator (ELES), TALUM and Jožef Štefan Institute) is cooperating with the US company Sargent & Lundy to prepare a Pre-feasibility study on building SMRs in Slovenia. It is planned that the report will be finished by the end of the year 2024. GEN energija's successful application for partnership in the European Industrial Alliance on SMRs indicates

new opportunities for the company to help facilitate and accelerate the development, demonstration, and deployment of SMRs in Europe by the early 2030s.

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Challenges in Modelling of Passive Heat Removal Systems for Small and Micro Modular Reactors

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Nuclear power plants (NPPs) will be integrated into a different energy supply system, necessitating dynamic operation and the provision of a diverse range of energy services, such as secondary regulation and supply of hydrogen and high temperature heat – tasks that nuclear has not undertaken previously. Conventional NPPs with large reactors find it very difficult to respond to all these challenges. Therefore, the development of innovative advanced reactors is crucial to achieving climate goals. These reactors could be of different sizes, but from the perspective of providing new energy services, small (< 300 MWe) and micro units (< 10 MWe) seem to be the most promising.

Innovative nuclear technologies differ significantly from conventional NPPs. A large share of advance reactors is designed to operate at much higher temperatures (500 to 1100 °C) than the conventional light water reactors (around 300 °C). Individual components of the primary circuit are typically integrated within the reactor vessel. The heat is often transferred to secondary system with novel heat exchangers, such as compact plate (NUWARD) and helicoil steam generators (NuScale), along with elongated heat pipes (eVinci). Furthermore, the innovative passive systems for the residual heat removal are usually considered in reactor design. These systems (among others) include Stirling engines, supercritical carbon dioxide loops, and heat pipes with liquid metals.

Given the significant differences in design and operation of advanced small and micro reactors as compared to conventional NPPs, the validity of existing system codes is challenged, making it crucial to develop new approaches in system modelling. Many of the forementioned characteristics are currently still beyond the validation scope of established system codes like RELAP5, ATHLET, APROS, or TRACE. This necessitates the development of novel constitutive models that can accurately predict the performance of advanced reactors under various operational conditions, including transient and accident scenarios.

This paper presents a comprehensive overview of the current state of system modelling for Small Modular Reactors (SMRs) and Micro Modular Reactors (MMRs), with a particular focus on innovative passive heat removal systems, such as heat pipes and supercritical carbon dioxide loops. The objective is to identify the most significant deficiencies in the constitutive models utilized in system codes and to propose directions for our future research.

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New reactor designs and small modular reactors

Nuclear jumps into space exploration challenges

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Framatome is an international leader in nuclear energy recognized for its innovative, digital and value added solutions for the global nuclear fleet. With worldwide expertise and a proven track record for

reliability and performance, the company designs, services and installs components, fuel, and instrumentation and control systems for nuclear power plants. Its more than 18,000 employees work every day to help Framatome's customers supply ever cleaner, safer and more economical low-carbon energy.
Space industries are facing new challenges as lunar long term settlement or reduction of Mars travel. They are looking for a safe, reliant and continuous energy enabling this ambition. Nuclear energy should be part of the solution.
Framatome Space, alunched in October 2023, is a dedicated brand to adress Space application. The objective is to put Framatome 65 years of nuclear and industrial expertise at the service of the space industry, transforming research project into industrial reality.

In this presentation, we will present the role of nuclear energy in the history of space applications. We will address the new ambition and how Framatome can bring its knowledge and industrial capacity to allow human beings to explore our solar system and beyond.

Fuel cycle, radioactive waste, and decommissioning

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Fuel cycle, radioactive waste, and decommissioning

Status of Slovenian LILW disposal facility

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The project of low and intermediate level waste (LILW) disposal facility in Slovenia started with the siting process in a year 2004, five years later the disposal concept and location were approved by the government decree within the state spatial plan. Environmental consent was issued in 2021, in subsequent years the construction licenses for a nuclear facility and infrastructure were gained.

The construction itself was divided in three phases. The first phase is a construction of the infrastructure and provision of technical security. The work started in August 2023 with the road renovation, sewage, electricity and telecommunications. The physical security with the outside fence (around nuclear facility) was completed and is already in operation. The second phase is the construction of nuclear facility for which the contract was signed in April 2024 and the work will start in second half of the year. The third phase is manufacturing and placement of portal crane for which the tender process was published in March 2024.

The construction itself will take approximately a three-and-a-half-years. After the construction the trial operation will start and then operation till year 2030 following with a standby phase with no disposal until 2049 and then the remaining LILW will be disposed. Closing is envisioned for 2059 and the institutional control of 300 years will follow, with 50 years of active and 250 years of passive control.

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Fuel cycle, radioactive waste, and decommissioning

Radioactive Waste Disposal in Germany

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The porposed paper will discuss the current situation in Germany regarding the radioactive waste disposal, focusing on low and intermediate level waste (LILW).

It will start with an overview of the waste disposal that is already closed at the Morsleben site, and the authors will show which acitivies have been performed in the last years. It will continue with information about a site that was used as a disposal, but does not meet the preconditions anymore - Asse. We will present the current status of the site and show which measures will be done in the future to safely manage the waste further. It wil also show the efforts a waste retrieval is taking, by sharing information about the process from the decision making till the actual retrieval.

The last disposal site will be the future Konrad site, which is currently designated to be the central disposal site for German LILW. We will talk through the desicion process and show the steps that are necessary for the sucessful implementation of the disposal site.

The last section will briefly discuss the current status of the search for a disposal for high level waste.

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Fuel cycle, radioactive waste, and decommissioning

Exposing Iron-rich Inorganic Polymers to Cobalt-60 radiation: Leaching behavior of Cs, Sr, and Eu nitrates and structural changes of the material

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Society and the nuclear industry have serious concerns about the handling of radioactive waste. Liquid nuclear waste containing radionuclides poses particular challenges due to their long half-lives and potential for environmental harm. Research has shown that inorganic polymers (IPs), also known as geopolymers, can effectively immobilize radionuclides by forming stable structures that incorporate them in their matrix. These IPs can be produced by industrial by-products, like iron-rich slags, and have been gaining attention in the scientific community because of their ability to create sustainable construction materials. However, concerns persist regarding their long-term stability and the leaching behaviour of these structures. A promising alternative for the immobilization of nuclear waste is to incorporate radioactive waste containing radionuclides such as Europium (Eu), Caesium (Cs), and Strontium (Sr) into these IPs. This study explores the impact of gamma irradiation on IPs made from iron-rich slag and doped with Eu, Cs, and Sr nitrates simulating the presence of radionuclides. The samples were exposed to gamma radiation from a cobalt-60 (^{60}Co) source, to a total dose of 6500 kGy. Leaching tests were conducted to assess the release of radionuclides under non-irradiated conditions and after irradiation, while nanoindentation and SEM characterization were used to evaluate mechanical and microstructural changes, respectively. Results showed minimal effects of irradiation on sample hardness and no significant differences in the release of dopants or structural elements between irradiated and non-irradiated samples. This research aims to advance strategies for nuclear waste management and environmental protection by examining how radiation exposure affects material properties and leaching behavior.

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Fuel cycle, radioactive waste, and decommissioning

Validation of the improved Siempelkamp NIS 3D activation calculation method for nuclear decommissioning

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The decommissioning of a nuclear power plant is an essential and natural part of its lifecycle and poses considerable technical and economic challenges for the energy providers. For the decommissioning of a nuclear power plant, precise knowledge of the radioactive inventory of all components resulting from decades of plant operation is required. On the one hand, this is required for the decommissioning license, on the other hand, it is economically impracticable to carry out sampling measurements everywhere in the plant and for each individual component. For this reason, exact calculations of the radioactive inventory of all components are indispensable with respect to a safe and economic decommissioning planning.

In past years, Siempelkamp NIS Ingenieurgesellschaft mbH achieved this task by developing an innovative new calculation method [1-3] based on the 3D simulation software MCNP. In addition to the detailed realistic modelling of the 3D geometry, this new method utilized, for the first time, the entire irradiation history of the power plants over several decades of operation, with all inserted fuel elements modelled pin-by-pin. This combination of a high-accuracy neutron source with data from the operational history and sophisticated Monte Carlo modelling using MCNP made it possible to calculate the radioactive inventory of a power plant with high accuracy. This calculation method was applied successfully, also on an international level, to several nuclear power plants, both PWR and BWR.

In 2020, this calculation method was further improved by Siempelkamp NIS and was applied to a German BWR. The new improved calculation method was presented in [4]. This new improved Siempelkamp NIS 3D activation calculation method represents a further significant improvement both methodologically and in accuracy. The new method uses sophisticated MCNP modelling with the mesh-tally feature provided by MCNP: For every single mesh-segment of the power plant model (approx. 100000 meshes) the neutron flux, the full 252-group neutron energy spectrum, as well as the reaction rates (cross sections) of all relevant reactions are calculated. Then, for each individual mesh-tally the activation calculations are performed with the newest ORIGEN-S version (part of SCALE 6.2.4) for all radionuclides and specified decay times using the MCNP-calculated cross sections.

Thus, for the first time in the community, three top state-of-the-art calculational features are combined: First, a high-detail pin-by-pin neutron source based on the operational history, second, a high-detail mesh-based 3D MCNP Monte Carlo model with the neutron spectrum and reaction rates calculated by MNCP and third, an ORIGEN-S calculation for each mesh-tally with cross-sections from MCNP reaction rates for the most important nuclides and 252-g condensed cross-sections for all other isotopes. This results in huge data volumes but also to activation calculations with unprecedented accuracy. Until today, the new improved activation calculation method was further applied to several German NPP units, both BWR and PWR.

An essential part of the decommissioning of an NPP is the packaging planning of activated NPP components. For this purpose, the activation calculation method must be validated by measurements. Here a combination of several experimental methods is very useful.

In the course of the ongoing decommissioning of several German NPP, an extensive measurement campaign was implemented by the energy provider. The measurements included essential components of the NPPs like the RPV, core shroud, upper & lower core grids, steam separators, steam dryers, the RPV bottom & head, biological shield, etc., using several measurement methods and detecting most of the important radionuclides. These extensive measurements allowed for the comprehensive validation of our activation calculation method which is presented in this paper for the first time.

The activation calculations show excellent agreement with the measurements up to several percent, thus representing the most accurate activation calculation method currently available.

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Fuel cycle, radioactive waste, and decommissioning

Automated object detection and position extraction of legacy nuclear waste using a robotic manipulator

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Nuclear waste comes in four main categories, high, intermediate, low, and very low-level waste. In the UK there are several legacy waste piles containing a variety of possible objects. Some objects are consistently similar enough that they can be categorized collectively. One such example is Magnox fuel element debris (FED), particularly consisting of hulls and springs from fuel elements that were cast into waste piles throughout the second half of the 20th century [1]. These objects are often highly radioactive. They need to be extracted from the pile, so they can then be disposed of appropriately. Automation of this extraction requires identification of object spatial locations.

Machine learning object detection is a method of utilizing neural networks with convolution across sets of pixels to extract patterns from images and use combinations of such patterns to recognize objects based on training via exposure to large amounts of similar images of similar objects. Yolo (you only look once) is an algorithm designed by ultralytics to efficiently perform object detection [2]. The bounding boxes an object detection algorithm draws onto an image when performing a prediction consist of coordinates defining the box's corners or centre, which can be used to extract spatial information if the distance from the camera is known.

Robotic manipulator arms with seven degrees of freedom (DOF) can reach any position within their working envelope at any orientation, such that versatile rotation and motion can be performed around objects with complex topography if the spatial location of the object is known in relation to the robot [3].

YoloV8 was trained on images of mock FED waste. A camera was attached to the end effector of a seven DOF robotic manipulator and the trained Yolov8 model was used to predict bounding boxes on images taken with it and the box coordinates were extracted. Range finding techniques were then used with these coordinates to determine spatial locations of the objects in relation to the world base of the robot, enabling precise automatic robotic movement to the object centre at a vertical standoff distance.

The spatial locations of the objects can then used to perform photogrametry techniques and produce labelled 3D models of the objects as part of a combined automated process and data stream. This can then be combined with gamma spectrometry techniques to identify localised radiation hotspots and extract activty levels [4].

Young author award contribution: Alex Carpenter organised the acquiring of mock data objects from industry partners with supervision from Thomas Scott. Alex captured the training data images, trained the YoloV8 algorithm, connected the cameras to the robot arm, and set up the data pipeline to allow the algorithm to perform predictions. Alex also wrote the code to perform all robotic movement, and process the data from the cameras and algorithm, with supervision from David Megson-Smith.

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Fuel cycle, radioactive waste, and decommissioning

Results from Knowledge Management activities in the EURAD programme

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In EC co-funded EURAD programme (European Joint Programme on Radioactive Waste Management, 2019-2024) particular emphasis was dedicated to Knowledge Management (KM) activities to ensure the capture of existing knowledge, transfer of knowledge between Members States (advanced and early-stage programmes) and management of the knowledge for future generations. Within EURAD there were a variety of tools and methods that support knowledge management activities with dedicated work packages (WP). In WP1 focus was on EURAD roadmap activities to systematically orientate users to existing knowledge and needs for research and technology development via a generic roadmap for implementing radioactive waste management, from generation to disposal. The Roadmap provides an integrated and systemic framework for organising, structuring and sharing available RWM knowledge in different levels, including themes, sub-themes and domains. Within WP11 State of Knowledge was collected with experts' view of the most relevant knowledge and associated uncertainties in a specific domain of the roadmap applied in the context of a radioactive waste management programme. WP12 Guidance offered activities consisting of developing a comprehensive suite of instructional guidance documents that can be used by Member-States with RWM programmes. In WP13 Training and Mobility activities consisted of developing a diverse portfolio of tailored basic and specialised training courses taking stock of and building upon already existing initiatives and creating new initiatives to bridge the identified gaps. The paper will present the current achievement in EURAD KM activities and plans for future work in EURAD2 partnership, projected to start in October 2024.

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Wall Crawling Robot for Nuclear Environment

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Working at elevated heights poses a substantial risk to workers, accounting for 30% of work-related fatalities in the UK during 2022/2023[1]. To mitigate these hazards, cutting-edge robotic systems are being developed to reduce the number of tasks performed at elevated positions. Nuclear Restoration Services (NRS) is executing a rolling programme to decommission the first generation of Magnox sites. This approach ensures that insights gained from dismantling plants are integrated into future site plans. The programme encompasses the decommissioning of 10 Magnox commercial reactor sites (22 reactors), with an estimated cost of £7.5 billion. Following decommissioning, these sites will enter a 'care and maintenance' phase, where they will be secured to allow radiation levels to decay[2]. A critical aspect of mitigating the risks associated with working at heights involves the inspection and characterization of large boilers, which is essential for determining their waste disposal routes. At Dungeness, boilers are situated in a compact radiological environment, necessitating the use of costly scaffolding for workers to collect samples. To reduce unnecessary radiation exposure and the inherent risks of working at heights, while significantly cutting costs, a bespoke wall-crawling robot has been developed. This advanced robotic platform employs permanent magnets for adhesion and is equipped with sophisticated sensor technology to enable precise odometry on vertical surfaces. The robot integrates odometry data with radiation measurements from a gamma spectrometer, facilitating the identification of radiation hotspots and the creation of detailed radiological maps. References [1] HSE (2023). Statistics - Fatal injuries in Great Britain. [online] Hse.gov.uk. Available at: https://www.hse.gov.uk/statistics/fatals.htm. [2] [2] Comptroller and Auditor General, National Audit Office, "Progress report: Terminating the Magnox contract," The Nuclear Decommissioning Authority, HC 727, Session 2019-2021, 11 September 2020

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Investigation of the Use of Reprocessed Uranium Fuel in VVER-440 Reactors

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Slovakia currently operates 5 units of VVER-440 reactors with one more being constructed. Over the past 40 years, these reactors have used fresh uranium fuel, starting with unprofiled type with 3.6% enrichment, through the current design of profiled fuel assemblies with burnable absorbers and an average enrichment of 4.87 %. The next step in improving the fuel cycles of our VVER-440 units is the use of a new fuel with 4.7 % enrichment, which will consist of a mixture of fresh and reprocessed uranium. This paper presents the first analysis carried out at the Slovak University of Technology in Bratislava investigating the use of this reprocessed fuel and its impact on integral and differential parameters during burnup, such as Keff, linear power, and decay heart. The analysis was carried out at a fuel assembly level using the SCALE6 system.

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Encapsulation of enhanced waste from Molten Salt Oxidation in geopolymer matrix

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During the operation of nuclear facilities, a high volume of Radioactive Organic Waste is generated. High prices for repositories force Radioactive Waste (RaW) producers to lower the volume of organic waste, and several processing techniques are being developed. The processes range from the stabilisation of Radioactive Solid Organic Waste (RSOW) in the geopolymer matrix up to thermal treatment like incineration or newly developing Molten Salt Oxidation. During the MSO process, the solid organic waste is dosed, together with an oxidising medium, under the surface of the molten salt, where flameless oxidation occurs, and non-combustible materials are trapped in the molten salts. The molten salts can vary for each process, but Na_2CO_3 is mainly used for its low melting point, good obtainability, and low corrosion medium for the reactor components. During the previous experiments, the molten salt waste was found to be very hard to stabilise due to its water absorption and formation of hydrates. This feature caused an efflorescence and cracking of the samples with higher waste loading and in contact with water. To improve the encapsulation properties of the samples, the molten salts were chemically enhanced through a chemical reaction with a solution of $Ca(OH)_2$ to form a more stable form. After the reaction, the decantate was mainly composed of $Na_2Ca(CO_3)2.5H_2O$. In the previous experiments, the resulting samples had high mechanical strength but low workability and fast setting time, which prevented the use of more than 15 wt.% waste loading. Therefore, the recipe was improved by adding 5 wt.% of water into the mixture to improve the workability, and the mixing time was reduced from 10 to 5 minutes. The curing of the prepared samples was under controlled conditions, and the XRD analysis and mechanical stress test were performed to determine their physical and mechanical properties. The experiments were conducted with 5, 10, 15 and 20 wt.% of enhanced waste added into the matrix. Compressive strength tests demonstrated satisfactory mechanical performance for future use, but more research is needed to determine possible waste load increases and sample stability.

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Radiation dose rate analysis of conceptual solution for the Croatian low- and intermediate-level radioactive waste storage

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In this paper radiation dose rate analysis is presented for the Croatian storage in which half of the low- and intermediate-level radioactive waste from the NPP Krško will be stored. It is of interest to determine the dose rates at locations where personnel is expected to spend some time during regular activities, both inside and outside the building. The calculations are performed using MCNP6.2 code. An MCNP model of the storage building conceptual solution with two main parts, i.e. storage area and reception/manipulation area has been developed. In the storage area there are six locations where reinforced concrete containers (RCC) will be placed in up to three layers. The corresponding volumetric gamma radiation sources are homogeneously distributed and uniformly sampled. Depending on the final decision, the containers can host four or eight metal drums containing radioactive waste (RCC4 or RCC8). The RCCs have an average isotopic activity. ANS/ANSI 1977 gamma flux-to-dose conversion factors and default cross section libraries are used. Multiple point detectors are placed outside the building and within the manipulation area to determine gamma ray dose rates. Additionally, a mesh tally is used to show gamma ray dose rate distribution and to investigate potential streaming paths. In addition to dose rates determination, the aim of this research is to quantify the skyshine effect as well as backscattering from the walls, since these effects might be the significant contributors to the total dose rate in case of efficient primary and secondary shielding.

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Development of irradiation and high-temperature resistant steels for nuclear applications

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In this work, we investigate new routes for the production of reduced activation ferritic-martensitic (RAFM) steels aiming to achieve specific improvements of their performance required for nuclear environment. Two specific challenges are addressed: (i) low-temperature embrittlement (LTE) and (ii) high-temperature creep (HTC) deformation. In this work, we review the optimization routes attempted to alleviate the above noted challenges which are otherwise met in European reference steel - EUROFER97. The development routes include the following approaches: (i) reduction of manganese and carbon content coupled with alternation of other chemical elements and followed by quench & rolling procedures; (ii) alternation of spatial distribution and morphology of carbonitrides by varying carbon, vanadium and tantalum content followed by heat treatment optimization; (iii) doping with zirconium/titanium and increase of tantalum content to improve ductility and toughness, without compromising strength and DBTT. The results of the baseline characterization including mechanical tests and various microstructural parameters are presented. The results available after first screening neutron irradiations and high temperature creep tests are presented as well. The achieved improvements are discussed and rationalized as well as lessons learned are summarized prompting the outlook for further investigations.

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Stress Corrosion Cracking of AM 316L in PWR Primary Water

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Additive manufacturing (AM) is one of the possibilities how the nuclear industry could tackle the problems related to replacement and design of complicated components thus to increase efficiency and safety of current water cooled as well as future nuclear reactors. The Horizon 2020 EU-funded NUCOBAM project main target was to develop the qualification process and evaluate the in-service behavior of AM materials, allowing thus in longer term use of additively manufactured components in current and future nuclear installations. One of the objectives was to evaluate and validate AM 316L austenitic stainless steel (ASS) in service performance. Therefore, focus of dedicated test program was on the assessment of the AM 316L properties as regards as the main relevant degradation phenomena such as corrosion, stress corrosion cracking (SCC), creep and thermal aging. In terms of evaluation of susceptibility to environmental caused degradation processes such as corrosion and SCC, focus was on performance of AM 316L in PWR conditions. In line with agreed project plan, JRC has carried out SCC susceptibility tests in modified PWR primary water. In the first phase, four slow strain rate tensile (SSRT) tests of four tensile specimens, with one specimen for 4 out of 16 different material variations) were conducted. Besides that, JRC also performed set of comparison SSRT tests for conventionally manufactured 316L ASS using the same strain rates. All tests were conducted in high temperature water simulating in terms of temperature, water pressure and chemical composition PWR primary water chemistry which was modified by increase of dissolved oxygen content. The tensile specimens were loaded with constant strain rate 1x10-7s^{-1} up to the fracture of the exposed specimens. After the tests, all exposed specimens were supplied to Fractographic Laboratory, Department of Materials, Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague. Three main objectives for the SEM fractographic analysis were the description of fracture micromorphology and the determination of failure mechanisms, assessment of the heat treatment impact on fracture micromorphology as well as observation of surface state with the focus on the presence of cracks outside

the main fracture area. Mechanisms of stress corrosion cracks propagation were very similar for material prepared by conventional technology and for material prepared by AM technology. Stress corrosion cracks propagated for both materials by combination of intergranular and transgranular crack growth. The cracks were found practically in the whole measured part of the samples due to geometry of tested specimens.

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Chemical production via radiolysis: chemical reactor design optimization

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Exposing a material to ionizing radiation splits its molecules or atoms to create charged species, free radicals and excited states [1], which later recombine and might even favorably modify the interface structure of a catalytic converter [2]. Essentially, radiation brings additional energy into the system that can be used to split tightly bound molecules [3] such as CO2+H2 to form ethanol. Optimizing the amount of energy that is deposited in the reactants plays an important role. While neutral particles such as incident neutrons and gamma rays rarely interact in a gas, it is common practice to try to transform them into charged particles, which deposit their energy very quickly. In this work we present a computational study on using metallic foams, submerged in the chemicals to study their effectiveness on the energy deposition within the reactants. We performed a scoping analysis, varying the foam's feature size and porosity. Secondary charged particles were also simulated. We note that tightly packed foam with low porosity and small feature size is the most effective in depositing energy in the reactants. For incident gamma-rays reaction volumes with characteristic sizes of a few centimeters and medium (50-60%) porosity of the metallic foam also shows similar performance, where 87%+ of the total energydeposition occurs in the reactants.

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The FFT-Homogenization Method in Irradiated Austenitic Stainless Steels

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Austenitic stainless steels are highly valued as internal structures within the light water reactor in nuclear power plants due to their outstanding mechanical properties, coupled with robust resistance to stress-corrosion cracking and irradiation damage. Despite these benefits, prolonged exposure to the harsh nuclear environment can lead to significant degradation of these properties. This deterioration is typically characterized by a noticeable loss of toughness and ductility, an elevation in yield stress, and a diminishing work hardening capacity, phenomena that have been consistently reported nowadays.

In response to these challenges, this study establishes the FFT-homogenization method within the crystal plasticity framework to meticulously examine the effects of irradiation on the irradiated austenitic stainless steel crystals. This approach considers various levels of neutron irradiation doses, as calibrated through experimental studies. Moreover, to ensure the reliability and accuracy of the method, comparisons with established finite element methods have been conducted. This rigorous validation process underscores the effectiveness of the FFT-homogenization method in simulating the complex behaviors of irradiated austenitic stainless steels, providing insights that are crucial for enhancing the longevity and performance of nuclear power plant components.

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Crack growth prediction with phase-field method

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Fracture and crack growth in structural components present a significant challenge to the lifetime of a nuclear power plant. Thus, the prediction of crack initiation, growth and consequently the component failure is important.

Some of the existing models and methods for crack growth prediction successfully replicate experimental data. However, their limitations include the set-up of predefined crack (growth) path geometry, dependence on several material constants that should be obtained experimentally and need to apply complex re-meshing strategies within finite element simulations. The phase-field method, which has been under development for the past decade, could help to overcome some of these limitations, as it can model the initiation of crack propagation and path in both brittle and ductile materials under different damage phenomena, such as pure fracture or fatigue.

In this paper, simple cases of current phase-field method implementations will be presented. Numerical simulations will be performed with the ABAQUS finite element method code. The simulation results representing relevant mechanical parameters such as displacements, stresses and strains, will be compared to semi-analytical results and other models.

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Sounding Out Separation: Numerical Investigation of Utilizing Ultrasound for Medical, Fission, and Fusion Isotope Enrichment

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The current state-of-the-art techniques in nuclear engineering and medicine require materials enriched in a particular isotope. This ranges from medical diagnostic and therapeutical applications, serving some 50 million people (about the population of Italy) annually, fission fuel enrichment, producing some 10% of global electricity needs to research in the field of fusion (breeding blanket, deuterium production) and fission (chlorine molten salt reactors, zinc separation). Nowadays, most of the isotope separation is performed in large gas centrifuge facilities, where several stages of separation are cascaded together. The method works by establishing a density gradient in a gaseous medium via centrifugal forces, forcing molecules with heavier isotopes to move towards the outer centrifuge wall and be collected.

We propose an investigation into isotope separation techniques, where the centrifugal force is replaced by an ultrasonic transducer and a resonant cavity, establishing a standing wave, where the heavier isotopes would collect in regions of minimal pressure (nodes).

With the constant improvement in computer technology and the availability of general-purpose compute frameworks on consumer grade graphics processors, it is nowadays possible to simulate the behavior of a large collective of gaseous molecules with various boundary conditions.

The project aims to computationally analyze the behavior of gas molecules composed of different isotopes within a standing wave cavity. Initially, the focus will be set on developing a high-performance simulation code for studying the behavior of the mentioned gases in 2D, with investigations into various gases and gas mixtures. Subsequently, the simulation framework will be extended to 3D. Depending on the findings, there's potential to expand the framework to simulate liquid behavior by incorporating additional potential interactions between individual gas molecules.

Results of the above-mentioned research will demonstrate the practicality of isotope separation through ultrasonic waves, potentially laying the groundwork for designing and deploying a prototype enrichment device.

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Experimental Simulation of Harsh Radiation Environments through High-Energy Helium Implantation: Insights from Positron Annihilation Lifetime Spectroscopy

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This abstract presents the pioneering results of a novel high-energy (up to 17 MeV) and high-fluence (5.42 × 10^17 cm^−2) helium implantation experiment targeting ferritic/martensitic steels essential for nuclear applications. The experiment was aimed to achieve a quasi-homogenous displacement damage profile extending up to 70 micrometers within diverse sample sets. Positron Annihilation Lifetime Spectroscopy (PALS) was employed for thorough characterization of the implanted samples. Notably, this study explores the suitability of fitting models derived from previously published PALS experiments on similar materials irradiated in spallation neutron targets, affirming the viability of this methodology for experimental simulation of severe radiation environments. By encompassing a broad spectrum of helium concentrations within the matrix (ranging from 230 appm to 1825 appm), the investigation facilitated a comprehensive understanding of radiation-induced helium-vacancy clusters in terms of their size and He/V ratio. These findings contribute significantly to advancing the comprehension and simulation capabilities of radiation-induced phenomena in materials destined for demanding nuclear environments.

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Agglomeration and amorphous transformation of nanocrystalline silicon carbide (3C-SiC) particles under the irradiation

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Over the past few years, SiC has been studied widely due to its absolute physical and chemical properties. Silicon carbide is properly applied as an attractive semiconductor which has high melting point and high physical - chemical stability. According to all mentioned properties of SiC has absolute application potential in fuel reactors and other nuclear technologies. When materials are applied in nuclear reactors or technologies they are exposed to ionizing radiation. The study of changes occurred in the structure of substances or defects are essential. Nanocrystalline 3C-SiC have wide application potential on the fusion-fission reactors due to its attractive physical properties. High oxidation and radiation resistance at high temperatures increase application potential of 3C-SiC nanoparticles. The effect of neutron irradiation on silicon carbide has been studied in several researches up to the present day [22-24]. Different types of defects occurred in 3C-SiC nano crystal as a result of effect of neutron flux cause changes its physical properties. However, the nature of these defects hasn’t been studied properly until today.

It is clear from TEM and SEM analysis that nanocrystalline 3C-SiC particles agglomerated maximum 70-80nm. After neutron irradiation there is thicker (approximately 3 nm thick) amorphous layer surrounding the 3C-SiC nanoparticles. Amorphous layer of the surfaces of nanoparticles can cause a greater or lesser agglomeration degree. Higher magnification at the TEM device existed atoms were observed in crystalline lattice and nanocrystalline nature of material was approved. Defects or clusters formed in nanomaterial after neutron irradiation were explained by “spots” observed in TEM images. It is clear from EDP analysis that neutron irradiation doesn’t affect crystal structure of 3C-SiC nanoparticles.

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Oxidation Behaviour of Accident Tolerant Fuel Cladding Materials at High Temperatures

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Since the Fukushima accident, research and development of accident tolerant fuels (ATF) has accelerated. The development of ATF cladding materials aims to further increase the safety of nuclear power plants. Zirconium fuel claddings used in water-cooled reactors can oxidize in steam under accident conditions and hydrogen can be evolved during oxidation. The amount of hydrogen generated in an accident situation can be significantly reduced by forming a coating on the surface of the zirconium cladding that has extremely good corrosion resistance (e.g. chromium). In order to obtain more information on the oxidation behaviour of ATF cladding materials at high temperatures, a new test series have been initiated at the HUN-REN Centre for Energy Research. The test series was carried out with different types of coated (Cr-, CrN-, CrN/Cr-, TiAl-coated) and uncoated zirconium alloys (Zry-4, Zirlo, Opt. Zirlo). Non-zirconium cladding samples (FeCrAl) were also oxidized.

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A review on fuel swelling of oxide fuels

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At high burn-up the change of volume is dominated by the fuel swelling driven by the precipitation of the insoluble fission products in the fuel matrix, which is either in solid or gaseous state. Insoluble solid fission products that segregate in the form of metallic or oxide inclusions, or noble fission gas atoms that remain trapped in the fuel matrix, are associated with atomic volumes that vary respectively between 1.3 and 2 times that of the U atoms in the host UO2 lattice. Fuel performance such as heat transfer at the fuel-cladding gap (gap closure) and cladding strain are correlated with fuel swelling.

In this paper we review results on this quantity giving great attention to recent findings. Models employed in fuel performance codes are presented and discussed.

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PTS analyses of a PWR with cracks during an SB-LOCA event with consideration of LTO improvements

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During the operation of a pressurized water reactor (PWR), it is important to ensure that existing crack-like flaws in the reactor pressure vessel (RPV) will not propagate under brittle fracture during emergency events that may lead to a pressurized thermal shock (PTS), such as loss-of-coolant accidents (LOCAs). This is even more crucial as nuclear power plants enter long-term operation (LTO) beyond the designed operational lifetime. For a selected transient event, a PTS analysis involves a thermo-mechanical analysis of the RPV to evaluate the temperature and stress distributions through the vessel's wall, followed by fracture-mechanics analyses to obtain stress-intensity factors (SIFs) of postulated cracks, which are finally compared with the fracture toughness of the RPV material to determine the likelihood of brittle fracture.

In this paper, PTS analyses of a reference small-break LOCA (SB-LOCA) due to a 50 cm^2 break in the hot leg with concurrent loss of offsite power are performed on a 4-loop Kraftwerk Union KWU-1300 PWR – Konvoi German design – plant. Additionally, PTS analyses are also carried-out when the same SB-LOCA event occurs in the plant with three LTO improvements implemented separately, which a-priori have beneficial effects for PTS. The LTO improvements include (i) LTO-1: heating of water in high-pressure injection system tanks from 15°C to 45°C, (ii) LTO-2: heating of water in accumulators from 20°C to 50°C and (iii) LTO-8: change of secondary-side cooldown rate - operator action - from 100 K/h to 200 K/h.

In all the studied cases (reference and LTO improvements), the analyses are performed with the FAVOR code as well as with an in-house implantation of SIF solutions developed by CEA. For the reference case and LTO-8, the analyses are additionally performed with a 3D finite-element model of the RPV, together with 3D fracture mechanics submodels of a small portion of the RPV with postulated cracks. Axially and circumferentially oriented through-clad cracks (TCC) and embedded cracks in the RPV wall are considered in the PTS analyses assuming the

ASME and Master curve as a fracture-toughness concepts, as well as the Tangent and maximum warm pre-stress approaches for onset of crack initiation. The goal of the paper is to analyze whether the LTO improvements yield considerable gains (margins) in terms of maximum allowable adjusted reference temperatures as compared to the reference case. This work has been performed in partial fulfillment of the European project APAL (Advanced PTS Analysis for LTO).

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Evaluation of Average Natural Background Radiation Dose in Slovenia

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Natural ionizing radiation contributes a major share to the dose a person receives. The sources of ionizing radiation can be grouped into different categories. In the first group there are high-energy cosmic rays hitting the Earth's atmosphere and releasing secondary radiation, while in the second group there is radiation from radioactive nuclides that are present from various nucleosyntheses in the Earth's crust since the formation of our planet. Due to different geological formations of surface bedrocks, the radionuclide concentrations may vary between continents, regions or countries.

The value of the annual dose due to natural radiation is of great importance as it can be used for contextual comparison to the radiation exposure in medicine or to the dose to the general public in the event of a radiation or nuclear accident.

Historically, the earliest estimate of the annual dose received by a Slovenian citizen dates to 1989 as part of a study to determine the additional exposure of people living near the Krško nuclear power plant in Slovenia. The annual dose from natural radiation was determined to be 2.4 mSv, half of it from radon exposure and the rest from cosmic radiation and radioactive isotopes in soil, food, water and air. Further measurements were carried out in the following years, but the total dose assessment has remained the same to this day.

Since the first assessment of the annual dose received by a resident in Slovenia, numerous measurements of radon concentrations in buildings in Slovenia, concentrations of radioactive isotopes in soil, food, drinking water and air and doses in buildings due to natural radiation have been carried out. In addition, Slovenian legislation on radon and ionizing radiation was following international recommendations and was hence amended in 2018, resulting in new dose conversion factors and concentration limits. For this reason, a new assessment of the annual dose rate due to natural background in Slovenia was performed and is estimated to be ~ 6 mSv.

This paper presents the reassessed contributions to the dose from natural background radiation. The largest contribution to the reassessed dose is radon, which accounts for almost 83% of the dose, while the other components have remained almost the same as in 1989. The large contribution of radon to the dose is due to the higher average radon concentration in Slovenian buildings, as first measurements in 1995 have shown. In addition, international studies have shown that the relationship between radon exposure and dose estimation was under evaluated in the past, so the dose factor was increased by two.

The study also assessed for the first time the contribution of radioactive isotopes in building materials to the dose received by a person, as this contribution was previously included in the radiation of radioactive isotopes in soil.

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Radiation and environmental protection

The Utilisation of Legacy Mine Sites as Training Environments for Radiological Emergency Response Preparedness

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After events such as those of Chornobyl in 1986 and the Fukushima Diachii Nuclear Power plant accident in 2011, the International Atomic Energy Authority (IAEA) has introduced guidelines to ensure preparedness for nuclear and radiological incidents that are all focused on ensuring the protection of people and the environment from the harmful effects of ionising radiation [1]. Effective preparedness hinges on comprehensive and routine training regimes that simulate the high-stress conditions of radiological incidents.

It is estimated that the United States alone has approximately half a million legacy mine sites distributed across nearly all 50 states, the overwhelming majority of these are deemed abandoned and in need of further remediation [2]. These mines were once extracted for valuable commodities such as gold, silver, copper, lead and even uranium. If the mines weren't directly exploited for uranium themselves, the governing geological processes that form the precious metalliferous minerals and ores often have accessory naturally occurring radioactive material (NORM) coincident with them. Since these are often found in trace amounts, at the time of the mining were not considered economically viable to extract and so were often discarded in spoil heaps. The result of this is mine spoil also known as tailings containing radioisotopes and heavy metals that have been left onsite or near the disused mine site. This often leaves radiological 'fingerprints' around the site that can be localised, characterised and mapped [3], [4].

Legacy mine sites, with their unique and complex infrastructures, present ideal environments for conducting realistic and rigorous training simulations for radiological accident responses. These sites often feature underground tunnels, varied geological formations, and areas contaminated with NORM, which mimic the conditions that might be encountered during a nuclear disaster. This resemblance makes them excellent venues for emergency response teams to practice navigation, contamination assessment, decontamination processes, and emergency evacuation under realistic conditions. Moreover, the utilisation of robotics such as Uncrewed Aerial Vehicles (UAVs), Uncrewed Ground Vehicles (UGVs) and Subaqueous Remotely Operated Vehicles (ROVs) provide the ability to gain valuable insight into the target area and or environment from a safe distance.

This research showcases two field deployments to two legacy Uranium mine site locations, South Terras in the South-West of the UK and Lucky Boy Uranium Mine in Arizona, USA. These field deployments aimed to demonstrate the use of commercial off-the-shelf (COTS) robotics augmented with visual and gamma sensors to localise, characterise and map the sites from airborne and ground-based means for the application in nuclear response scenarios. With the use of photogrammetry and or 3D scanning Light Detection and Ranging (LiDAR) it is possible to 'fuse' the scintillator spectrometer data to develop a 'digital twin' of the site. This allowed us to remotely examine the radiological and structural safety of the legacy mine sites, which perfectly emulates a radiological response scenario demonstrating these otherwise abandoned features, as an ideal training ground.

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Radiation and environmental protection

Preliminary Results of the Dudváh River Contamination from the 1977 A1 Nuclear Reactor Accident

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The A1 nuclear power plant, part of the Jaslovské Bohunice site in Slovakia, experienced a significant incident on February 22, 1977. The reactor used was a Czechoslovak-designed KS-150 Gas Cooled Heavy Water moderated reactor. The blockage of the flow of the moderator between the fuel pins led to a partial meltdown and severe contamination of the primary loop with fission products. Due to leakage from the steam generators, the secondary loop was also contaminated, resulting in the discharge of radioactive materials into the environment. This first contaminated the Manivier drainage canal and subsequently the Dudváh River.This paper presents the initial results of an intended series of gamma spectrometric measurements of soil samples collected from the banks of the Dudváh River. The results presented are from soil samples gathered approximately 7.5 km downstream from the secondary loop outlet, in the village of Trakovice. The focus of the study is on the detection and quantification of radioisotopes released during the 1977 accident at the A1 nuclear reactor. Soil samples from various locations along the river were analyzed using NaI(Tl) detectors to assess current levels of contamination. The results show elevated concentrations of Cs-137 in the samples collected near the stream compared to background levels, or samples gathered farther from the banks, suggesting persistent environmental contamination from the historical reactor accident. These preliminary results are crucial for developing the methodology for future measurements, which will create a contamination map of this segment of the stream. The aim is not only to better understand the long-term environmental impact of the A1 reactor incident but also to establish a new reference point for contamination monitoring in case of similar accidents.

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Radiation and environmental protection

Calibration of HPGe detectors and usage of prompt gamma rays to extend detector efficiency curve in >2 MeV energy range

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This study refines the calibration method for high-purity germanium detectors (HPGe) using prompt gamma radiation from an irradiated Fe-56 target, with the aim of improving the accuracy of high-energy gamma-ray spectroscopy for applications such as the KATANA water activation loop. Traditionally, calibration with sources such as Americium-241 and Europium-152 has achieved a detector efficiency of up to 1.4 MeV. To increase this range and precision, prompt gamma rays from Fe-56 bombarded with high-energy neutrons were used, enabling precise efficiency measurements up to 8 MeV.

The experimental setup included a standard HPGe detection system optimized with shielding to ensure high-resolution gamma-ray detection. This refined method significantly improved calibration accuracy and achieved consistent and reliable calibration over a broad energy spectrum. The prompt Fe-56 gamma-ray results were integrated with standard source data and closely aligned with the MCNP simulations.

The development of this improved calibration technique is crucial for special applications such as the KATANA water activation loop, where accurate and precise measurements of high-energy gamma rays are essential. The improved detection capabilities of the calibrated HPGe detectors are particularly beneficial in environments that require detailed analysis of the interactions of high-energy gamma rays with water, which is critical to understanding neutron activation and other related processes.

This research is significant because it provides an improved calibration method that enhances the precision of HPGe detectors and expands their application in various high-energy fields of nuclear physics, astrophysics, and environmental monitoring. By refining the calibration of detectors specifically for projects such as KATANA, this method supports more detailed studies and contributes to improving the safety and operational efficiency of water activation systems.

Furthermore, this improved calibration approach contributes significantly to a better understanding of gamma-ray interactions and detector responses at high energies and provides a valuable reference for ongoing research and development in detector technology and activation analysis. By improving the calibration capabilities of HPGe detectors, this method facilitates advances in nuclear safety, reactor design and environmental protection, thus enriching the tools available to study complex nuclear environments.

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Radiation and environmental protection

The Dark Star System Architecture

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Stand-off detection of radioactive materials is of interest in the nuclear industry to locate contamination without bringing humans into the close proximity that is usually required for traditional Geiger-Müller detectors to be effective. Stand-off detection techniques have the potential of covering larger areas faster, with lower radiation doses incurred to the survey team. Of particular interest in the range of optical detection techniques is fluorescence imaging. Radioactive compounds containing uranyl or plutonyl ions may fluoresce in the visible wavelength range when excited by higher energy photons, depending on the presence of other ions in the molecular structure. This can be used to detect small quantities of oxidised uranium or plutonium bearing radioactive contamination on surfaces several metres away. The fluorescence intensity is proportional to the excitation energy, which means higher energy and/or incident flux results in stronger excitation. The Dark Star was designed and developed at the University of Bristol as a stand-off fluorescence imaging device for radioactive materials on surfaces. It uses an array of ultraviolet light emitting diodes at 365 nm to stimulate fluorescence, and capture the fluorescence with a sensitive camera. This poster presents the engineering design of the Dark Star and its system architecture, including component selection, power supply and management, and structural design considerations. The selection criteria for UV LEDs as a function of thermal management and excitation energy are discussed, feeding into system integration. Experiences deploying an early prototype in a radioactive environment and capturing preliminary data in a real-world use case scenario are also discussed.

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Radiation and environmental protection

Calculation of lead activation during cyclic exposure in LFR

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This work presents a simple one-dimensional model of a pool reactor to study coolant activation in Lead Fast Reactors (LFR) under prolonged exposure to high neutron flux. The coolant flows through the core and the other reactor components located inside the pool, undergoing cyclic exposure. The configuration is considered as stationary, with all the relevant components of the reactor represented by means of equivalent lengths compared to the flow passage. The flow speed is constant in each segment of the equivalent circuit, showing different levels of neutron flux along the loop. Periodic boundary conditions establish physical recirculation of the coolant. The model uses a Lagrangian approach to estimate the coolant nuclide inventory in the system, with the aim of providing an accurate prediction of the activity and dose rate after a long irradiation time. The simplification of the decay chain is discussed after identifying the most dangerous radioisotopes originated by decay and transmutation from the initial lead isotopic inventory. The presence of impurities in lead is also considered. Finally, a test case representing a typical LFR system designed by newcleo LFR is presented as example.

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Radiation and environmental protection

Automated Aerial Vegetation Mapping and Identification for Wildfires in the Chernobyl Exclusion Zone

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Wildfires are a near-annual occurrence in the Chernobyl Exclusion Zone (CEZ) located in North Ukraine, Pripyat, which is a designated 30 km alienation zone around the Chernobyl nuclear power plant. This zone was established shortly after the accident of April 1986 which occurred due to a malfunction of reactor unit 4 and led to a large release of radionuclides [1]. Since then, vegetation has taken over the exclusion zone due to stringent environmental regulations that have limited human activities such as forestry and farming[2]. Due to the lack of vegetation management the zone has an increased risk of wildfires which can migrate the radionuclides released during the accident outside of the exclusion zone. These radionuclides include Cs-137 and Sr-90 both of which are primarily located in the top first 5cm of topsoil [3] and have been absorbed by plants instead of sodium, potassium, and calcium [4, 5].

Traditionally wildfires have been detected using watchtowers, aircraft, and satellite imagery however these techniques are now being replaced with uncrewed aerial vehicles (UAVs). This imaging platform provides a real-time, high resolution and a risk free alternative for the pilot, this is especially crucial given the radiological threat of the smoke produced [6, 7]. Machine vision models have been applied in wildfire management and monitoring; these tools in combination with an inventory of vegetation and infrastructure maps can provide information to firefighters for effective control of wildfires [8]. Current models have used a satellite based detection and evolution system that allows for deployment of UAVs to the designated areas of the fire. The evolution of the fire is modelled using elevation data, wind patterns and the Normalised Difference Vegetation Index (NDVI). The NDVI is a measure of the reflectance of near infra-red (NIR) and red wavebands, which can be used to estimate the health, hydration, and density of vegetation[8, 9]. However, it provides an incomplete summary of the fuel distribution in the area, since it fails to differentiate between different vegetation species which can have wildly different burn rates. This is a particular issue in the CEZ since the absorption rates for radionuclides also depends on the vegetation species allowing two areas with a seemingly similar NDVI to pose very different levels of radiological threat [10]. Thus, additional information about the fuel distribution is required to improve the performance of wildfire simulations in the area.

In this work, a prototype vegetation monitoring and identification tool was developed using aerial images collected with a DJI Mavic 2 Pro. This tool consists of a machine learning model capable of semantically segmenting images into 7 different classes with (heath, shrubs, trees, grass, ice, snow, road) with an overall accuracy of 87.3%. It was trained on a dataset collected at Priddy Mineries (51°15'29.3"N 2°39'05.0"W) instead of the CEZ due to the ongoing Russian invasion of Ukraine. The location was selected for its similar terrain and vegetation as well as its high density of different vegetation classes. To our knowledge, a model accuracy of 87.3% is the highest scoring for any computer vision model aimed at semantic segmentation of vegetation in the CEZ and is 10.3% better than the only other model developed by M. Matsala et. al. in 2021 [2], however we have likely overestimated the model performance in the CEZ since it was bench-marked on images from Priddy Mineries. Thus, we cannot verify this until access to the CEZ has been restored.

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Radiation and environmental protection

The Slovenian Early Warning System For Radiation In The Environment

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In the event of a nuclear or radiological accident in the territory of Slovenia or abroad, one of the key tasks of the Slovenian Nuclear Safety Administration (SNSA) is to provide immediate data on radioactivity in the environment. These data are the basis for the successful implementation of protective measures for the population. Thus, it is important to establish and maintain a system of monitoring stations around the country as well as means of quickly assembling data and delivering them to the decision makers.

Soon after the Chernobyl disaster, Slovenia has set up the Slovenian Early Warning System (EWS). The EWS at the time consisted mainly of automatic dose rate meters, with later additions of three airborne radioactivity meters and two radioactivity deposition meters.

The EWS system consists of measuring devices, communication channels and software for collection, storage, display and alarming. The measuring stations, AMES MFM, were designed in the 90s, which means that they are technologically and functionally outdated, even though some improvements were introduced over the years. The last major investment in EWS was in 2006, when SNSA and the Slovenian Environment Agency (ARSO) implemented a project where the number of devices increased from then existing 19 to 54, and the first comprehensive software solution for data aggregation, alarming and storage was designed, unifying previously segmented elements.

Since 2017, SNSA is performing a large-scale renewal of the EWS, which is in the last phase, with plans to be completed by the end of 2024. The purpose of the project was to address and update all EWS aspects: data collection and display software, communication routes and measuring devices.

Based on experiences gained during the operation of the first aggregation software, SNSA designed the new central information system “Radioactivity in the Environment” (RVO, “Radioaktivnost V Okolju” in Slovenian). It is an automatic system for collecting, storing, and reporting dose-rate and activity concentration measurements in the environment. All measured data, as well as auxiliary information on the devices, are at the disposal of the SNSA staff. In addition to on-line measurements, off-line data on samples measured in the laboratories are also integrated. Simultaneous presentation of different measured values as time series, statistical and performance analysis, prompt detection of elevated values and subsequent alarming, are among key features of the system. Administration level access grant user the possibility to introduce new devices or expand off-line measurement databases. Additional modules enable a two-way communication with the mobile units and real-time data transfer from vehicle-based monitoring systems. The RVO was successfully implemented and begin functioning in 2019, yet it is under constant development with new features added each year based on operating experience.

The RVO is designed for both general public and SNSA staff use, providing prompt public access to dose-rate data, as well as access to the database of radiation measurements in the environment, alongside with other information about radioactivity in the environment that may be of interest.

After the completion of the new software, it was time to improve the measuring devices. The SNSA acquired a new dose rate meter, type Envinet MIRA, which was a donation from the International Atomic Energy Agency (IAEA). The RVO Portal´s software also had to be upgraded to enable import of data from the new instrument. Both, the dose rate meter and the RVO, successfully passed one year testing period.

Between 2021 to 2023, the SNSA successfully gained funds from the Ministry of environment and spatial planning to purchase 54 Envinet MIRA devices. Additionally, 10 devices were secured through donations from the IAEA, making the final number of new devices 64. During this period, intensive installation was carried out by the SNSA´s Monitoring Section, replacing the old measuring devices with new ones. By the end of 2023, the EWS consisted of 56 Envinet MIRA meters, with four new locations introduced in 2023. There are only two, logistically demanding, high-altitude locations remaining, where the installation is predicated on ensuring robust communication and surge protection, due to possible severe weather. In addition to new devices, 22 old MFM dose rate meters are still in function, 14 of which that are owned by nuclear facilities (13 in the vicinity of the NPP Krško, 1 in RC TRIGA, integrated in the EWS), with the rest temporarily serving as backups to MIRAs until the completion of the communication and electrical infrastructure at the sites.

New devices have several key features, lacking in the old MFMs. Two of the most important ones are enhanced sensitivity, ensuring quicker data gathering and faster alarming, as well as spontaneous communication, initiated by the device, when an alarm occurs. Additionally, redundant power supplies (electrical mains, battery and solar panel) ensure continuous operation while redundant communication channels ensure measured data reach the central gathering point at the RVO. The challenges encountered in the installation process as well as solutions and the gradual improvement of the overall performance of the new devices will be detailed in the paper. Long term maintenance and subsequent financial requirements will also be discussed, since it is often a weak point in the planning of such network and can lead to failure.

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Radiation and environmental protection

Diffusion neutron flux that formed due to the scattering of a narrow beam

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The attenuation law is valid for a narrow beam of particles. However, it cannot be applied directly to diffusion flux. Usually, in order to get adoptation to the diffusion flux, a correction factor called the build-up factor (BUF) is introduced into the formula of the attenuation law. However, such a correction is not sufficiently justified. This is evidenced by the multiple discrepancy of BUF values obtained in different studies.

Previous studies have shown that the attenuation of the diffusion flux is not an exponential function of the shield thickness. In addition, for a real assessment of the flux attenuation, it is necessary to take into account such an important characteristic of the diffusion flux as the angular spectrum. The angular spectrum of the diffusion flux is affected by the distribution of particle sources and the scattering characteristics of the layer in which the diffusion flux is formed.

The results of the study of the formation of the diffusion neutron flux when a narrow beam pass through the scatterer layer are presented. Computational modeling based on the Monte-Caro method and analytical generalization of the results were carried out. The angular spectrum for the considered problem was obtained and the influence of various parameters on it was analyzed.

Education

and training and public outreach

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Education and training and public outreach

International Higher Education with I2EN and the French Nuclear Industry

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The International Institute of Nuclear Energy (I2EN) is a prominent French institution that brings together major universities, engineering schools, and research organizations to advance global nuclear energy development. As a non-profit organization initiated by the French government, I2EN collaborates with ministries, the nuclear industry, research bodies, and educational institutions. Its mission includes representing French nuclear training internationally, evaluating and certifying programs, and maintaining a database of certified courses. I2EN addresses tailored training requests, leveraging its extensive network and expertise.

In this presentation, we will introduce I2EN's structure and mission, showcasing its diverse training programs, including higher education courses and professional training led by nuclear industry experts. We will highlight the experimental and operational facilities accessible through these programs, along with specialized training mock-ups for practical learning. Additionally, we will discuss I2EN's role in international networking, coordinating with European and global partners to facilitate student exchanges and internships, thereby enhancing global collaboration in nuclear education and training.

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Education and training and public outreach

Striving for Excellence: Improving Plant Systems Refresher Training at NPP Krško

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In November 1979, six months after the Three Mile Island (TMI) accident, the Nuclear Regulatory Commission (NRC) issued NUREG-0632, a document outlining the NRC's views and analysis of the recommendations from the President's Commission on the Accident at Three Mile Island. Among various recommendations, the Commission emphasized the need for comprehensive, ongoing training to ensure that operators maintain a high level of knowledge.

Today, nuclear power plants operate at full capacity throughout the fuel cycle without significant transient phenomena, which is advantageous for the power system and consumers. However, this steady operation does not necessarily benefit the operational crew in terms of experience. Maintaining a balance between optimal power system performance and crew experience is crucial. To prevent cognitive stagnation and to facilitate experience gain, simulators are employed. Each operator undergoes one month of training per year. But a closer examination of the training content reveals that refresher training on plant systems is not performed as often as it should be. The technical content is often overshadowed by soft skills and other non-technical topics.

This paper discusses how NPP Krško addressed the issue of infrequent refresher training on plant systems.

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Education and training and public outreach

Nuclear Technology Courses in Nuclear Training Centre Ljubljana

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The paper presents experiences from performing nuclear technology courses at the Nuclear Training Centre Ljubljana. There are two types of important courses, conducted for Krško NPP staff and other organizations, dealing with nuclear technology. The first course is called NPP Technology (the acronym in the Slovenian language is TJE) and is intended for future control room operators. This course is the first, theoretical part of the initial training of licensed operators (later stages – NPP systems and simulator training – take place at the location of the NPP). Approximately 5 months are devoted to different topics, such as nuclear and reactor physics, thermal-hydraulics and heat transfer, radiation protection, electrical engineering, materials, and nuclear safety. In 2024 the 20th edition of the course was conducted.

The second course, Basics of Nuclear Technology (in Slovenian OTJE) is suitable for other NPP technical personnel, technical support organizations, regulatory body, etc. This course consists of two parts: theory (4,5 weeks) and NPP Systems (3,5 weeks). In 2024 the 46th edition of the course was conducted.

The paper will present the course organization of the NPP Technology course, materials preparation, feedback from participants, and major improvements, that were implemented in the last years.

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Education and training and public outreach

Elevating Radiation Protection Training: Harnessing Visual Tools for Enhanced Learning

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Radiation protection training is crucial for ensuring the safety of those working with ionizing radiation. However, conventional methods often rely heavily on complex theories, making it challenging for learners to grasp practical applications. This article proposes a solution: integrating visual tools, specifically dose rate propagation programs, into training modules. These programs offer a dynamic learning experience by allowing trainees to visually explore how radiation spreads and how shielding affects it. Through interactive simulations, trainees gain a deeper understanding of radiation behaviour and the effectiveness of protective measures. This approach bridges the gap between theory and practice, empowering trainees with practical skills for real-world scenarios. By embracing visual tools, radiation protection training becomes more engaging, accessible, and ultimately, more effective in safeguarding workers.

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Education and training and public outreach

Youngsters about Nuclear Energy – Year 2024 Poll

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The Information Centre which is part of the Nuclear Training Centre at the Jožef Stefan Institute informs the visitors about nuclear power and nuclear technology, about radioactivity, about Krško Nuclear Power Plant and about energy in general.

Our main target population are the schoolchildren from the last grades of elementary school and from high school (ages 13-18) with their teachers. In the last decade we had close to 7000 visitors per year (except during limitations imposed by covid-19 pandemic). The visitors can choose between live lectures on nuclear technologies (fission and fusion), a lecture about use of radiation in medicine, industry and science and a lecture on stable isotopes. A general lecture about energy and an energy workshop is also available and usually performed for younger visitors. The visit includes a demonstration of radioactivity, a tour of our permanent exhibition and a virtual tour of the TRIGA research reactor.

Since 1993 we monitor the opinion trends by polling some 1000 youngsters. There are 10 questions in the poll and they remain unchanged for several years. This enables us to follow the trends in the basic knowledge of energy issues among youngsters and their attitude towards nuclear energy.

Nuclear fusion

and plasma technologies

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Nuclear fusion and plasma technologies

Construction and commissioning of the KATANA water activation loop at the JSI TRIGA research reactor

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Water as a primary coolant is used in most fission reactors today and will play an important role in the performance of fusion reactors. After being irradiated and activated, the cooling water flows through the cooling circuit, usually outside the primary biological shielding surrounding the reactor vessel, causing ionising radiation field throughout the facility. Consequently, additional protection and shielding for the instruments and personnel must be adequately considered. The threshold energy for the main activation reaction of water, i.e. $^{16}O(n,p)^{16}N$, is around 10.5 MeV. Neutrons in fusion reactors lead to a water activity that is 5 orders of magnitude higher than in fission reactors of similar power. Numerous computational analyses of the water activation process were carried out for ITER and DEMO. However, the understanding of cooling water as a radiation source is still insufficient. This is due to a lack of experimental nuclear data and inconsistencies between the main nuclear data libraries, inaccurate computational methods/codes that consider time- and spatial-dependent radiation sources (such as the flow of activated water in the cooling system), experimental facilities to validate the methodology, experimental facilities to calibrate gamma-ray detectors for higher energies and, above all, the lack of water activation experiments under fusion-relevant conditions.

Against this background, a closed-water activation loop, called KATANA [1][2], was built, successfully licenced, and fully commissioned in early 2024 at the TRIGA Mark II research reactor of the Jožef Stefan Institute (JSI TRIGA) in Slovenia. The KATANA irradiation facility will serve as a well-defined and stable 6 MeV – 7 MeV gamma-ray source and as a neutron source. Such a high-energy irradiation facility will enable various experiments based on water activation, e.g. shielding experiments with ITER-relevant materials, investigation of the response of detectors to high-energy gamma radiation, investigation of short-lived moving radiation sources, validation of computational codes/methods, etc.

The focus of this paper is on a comprehensive overview of the construction and commissioning phase of the KATANA irradiation facility. In particular, the cleaning of the radial piercing port, the construction of the inner and outer part of the closed-water loop, the construction of the concrete shielding wall in the form of a labyrinth, the safety systems, the installation of the electricity, the commissioning of the control room, the preliminary tests of the detector systems and the corresponding data acquisition system. Finally, the preliminary test experiments were also carried out with the KATANA irradiation facility.

The KATANA facility demonstrated the desired operating characteristics in the form of high and stable water flow rates, which led to high activity values of the observed activated isotopes $_{16}N$, $_{17}N$ and $_{19}O$, which is essential for minimising the experimental uncertainties. An extensive experimental campaign is planned for 2024 as part of the work package Preparation of ITER Operation in the framework of EUROfusion.

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Nuclear fusion and plasma technologies

Detection of defects and deuterium in displacement-damaged tungsten by ion beam methods in channeling configuration for fusion application

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Due to its advantageous characteristics, tungsten (W) is the main plasma-facing material candidate in future fusion reactors. However, in a future nuclear environment, the W crystal lattice will be heavily altered due to defects caused by 14 MeV neutrons from the D-T fusion reaction, affecting the physical properties of the material. To examine these defects created in the W lattice, we utilized Rutherford backscattering spectrometry in channeling configuration (RBS-C), a well-established method for studying lattice disorder and defect evolution induced by irradiation. To quantify the disorder, the change in the ion yield of light ions backscattered along a specific crystallographic direction is measured [1]. Moreover, when combining nuclear reaction analysis (NRA-C) with RBS-C, we can gain insights into the position of light species in the host material. In our case, we used the $D(^3He,p)^4He$ nuclear reaction to study the location of trapped D in displacement-damaged W.

To study the defects and trapping of D, we irradiated W (111) and (100) single crystals (SC) with 10.8 MeV W ions at two different doses and temperatures. Our goal was to create samples, with either dominantly single vacancies, small vacancy clusters or large vacancy clusters [2]. This was later on also confirmed by positron annihilation spectroscopy on our produced samples. The transmission electron microscopy (TEM) analysis of W (111) SC revealed dislocation lines and loops of different sizes, depending on the irradiation dose and temperature. Multi-energy RBS-C spectra analysis along the <111> direction unveiled distinct ion yield responses for each sample [3]. For the first time for W, we employed molecular dynamics (MD) simulations of overlapping cascades as input for RBSADEC code [4], used to simulate the RBS-C spectra. Simulated spectra, employing cells from MD, agreed remarkably well with experimental spectra for the lower dose sample, but showed discrepancies for the high-dose-irradiated sample, attributed to the presence of dislocation lines observed by TEM which cannot be formed in finite-size MD cells [3]. New simulation results with larger MD cell and different potential, giving better agreement for the high dose will be discussed.

Similarly, W (100) SCs were irradiated in the identical manner and defects were decorated with D. NRA-C and RBS-C spectra using a 0.8 MeV ^3He probing beam were measured to determine the D location in the tungsten lattice. The maximum signal was detected in the <100> axial and in the (110) planar channels. The interpretation of the spectra with the RBSADEC code[4], recently upgraded to simulate the NRA-C signal, will be discussed.

This work was preformed within EUROfusion Enabling research project ENR-MAT-01-JSI named DeHydroC.

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Nuclear fusion and plasma technologies

Comparative Analysis of SDDR Calculation Approaches in simplified Large-Scale Tokamak Models

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The shutdown dose rate (SDDR) in fusion reactors plays a critical role in the safety of personnel and the environment during maintenance and decommissioning phases, so understanding it is crucial. Accurate prediction and management of SDDR are essential aspects in the design and operation of fusion reactors.

In this study, SDDR is investigated in simplified models of large tokamaks to improve our understanding of the calculation methods and their performance. Two computational methods are analysed, the rigorous two-step method (R2S) and the direct one-step method (D1S). The JSIR2S code, developed at the Jožef Stefan Institute, combines calculations of the MCNP transport code and the FISPACT inventory code. It has been validated using measurements in the TRIGA research reactor and is now being evaluated for fusion applications. The results are compared with those of the D1S-UNED code, a one-step method that is used as a reference code for SDDR calculations in ITER and provides a basis for comparison with R2S codes.

This study aims to compare SDDR calculations at different cool-down times in fusion reactors using the above codes. The results will improve our understanding of SDDR phenomena and aid the development of larger fusion reactors like ITER.

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Nuclear fusion and plasma technologies

Analysis of four heat flux partitioning model at divertor cooling conditions

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In Eulerian two-fluid simulations, subcooled nucleate boiling on a heated surface is modelled as a boundary condition that determines the amount of heat that is partitioned between liquid heating and evaporative mass transfer. Mechanistic models achieve this partitioning though physically meaningful boiling parameters, the more important of which are: nucleation site density, bubble detachment frequency, and bubble detachment diameter. The classical three heat flux partitioning model, developed by Rensselaer Polytechnic Institute (RPI), was designed for operating conditions significantly different from those in the cooling loops of actively cooled divertor targets in fusion reactors, which are characterized by extremely high heat flux (10 MW/m^2), liquid subcooling (150 K), and flow velocity (10 m/s). Under such conditions, the RPI model predicts unrealistically high wall temperatures. Recently, researchers at the Massachusetts Institute of Technology (MIT) have proposed adaptations to this classical model. These adaptations include dependency of boiling parameters on local flow conditions and the inclusion of bubble sliding as a crucial mechanism of heat transfer in convective boiling flows. This study evaluates the capabilities of the four-heat flux partitioning model to simulate the boiling under conditions of high heat flux and high flow velocities.

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Thermo-Mechanical Study of Different Solutions for the Inner Top FW Module of DTT

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The primary objective of the divertor tokamak test (DTT) facility is to study the feasibility of innovative approaches to address the power exhaust challenges crucial for future fusion power plants. In this regard, it is anticipated that the first wall (FW) will absorb a substantial amount of heat during operation. Heat loads on the FW come from radiation and particle bombardment from the plasma inside the vacuum vessel (VV). The top FW (TFW) is fixed to the VV with supports. The module of the inner TFW includes a plasma facing component (PFC) attached to a backplate and support. There is a need to verify the suitability of the module's support so that the stresses resulting from electro-magnetic loads arising from plasma disruptions and thermal changes are not too high. Three design solutions have been proposed for the inner TFW module: plate design with plates as PFCs, coaxial design with coaxial flow pipes in tungsten monoblocks acting as the PFC, and the divertor-like design with tungsten monoblock PFCs.

This paper first describes the finite-element model development in ABAQUS of the three solutions of the inner TFW module. Then, these are employed in thermo-mechanical analyses that use thermal-hydraulic and electro-magnetic loads for the mechanical verification of the modules' supports. The results of the analyses include the displacements and stresses of the main components. These results are used in a comparative study of the three solutions (plate, coaxial, divertor) to evaluate their performance.

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Visualization experiment for analysis of two-phase flow in divertor cooling channels

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Dedicated flow visualization experiments are essential to gather key information for supporting the development of CFD closure models. Most of the experiments for investigation of convective boiling under realistic high heat flux conditions, relevant to fusion applications are very limited in their use of measurement techniques. Optical observations of the flow phenomena on opaque metal surfaces is not possible, high temperature and pressure conditions further hinder the use of accurate measurement methods. None of the available experimental studies at fusion-like conditions does not allow visualization of the convective boiling phenomena. To address this issue, a new experiment has been designed to visualize the boiling phenomena at these conditions. Through appropriate dimensional analysis, fusion-like conditions have been achieved using a surrogate working fluid that allows the observation of convective boiling in a transparent test section at much lower pressures and heating powers.

A novel visualization experiment consisting of a transparent test section included in a thermal-hydraulic loop filled with R245fa refrigerant is presented. The loop consists of a centrifugal pump, a Coriolis flowmeter for mass flow measurements, a heat exchanger for inlet temperature control, a transparent test section heated from the top, a condenser and a pressurizer. The boiling in the test section generates vapour bubbles which condense in the condenser. The pressurizer controls the pressure in the refrigerant loop. The heat exchanger and condenser are cooled by separate water-cooling loops. In this study, preliminary measurements on the new test section will

be described. Visualization will play a key role in the study of convective boiling. The boiling phenomena in the transparent test section will be studied using high-speed cameras in visual and infrared (IR) light. The matrix of experimental conditions will cover a wide range of heat fluxes, mass flow rates and liquid subcooling conditions. The experimental data obtained should provide insight into the boiling phenomena in the divertor cooling channels and shall be crucial for development and validation of appropriate theoretical models.

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Nuclear fusion and plasma technologies

Progress in fusion research by accelerating the Particle-In Cell code

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Fusion energy is a promising, clean and sustainable energy source. However, in order to realise its potential, the plasma dynamics in the fusion devices must be mastered. Particle-in-Cell (PIC) simulations, which simulate the behaviour of the plasma with unprecedented detail and thus help to optimise and predict the performance of the devices, are of central importance here. Fusion devices operate under extreme conditions and expose the plasma to intense heat and magnetic fields. Understanding the interactions between the plasma and the surface, particularly in the plasma sheath, is critical to the efficiency of the devices. PIC simulations provide a computational framework to accurately model these interactions and gain insight into phenomena such as material erosion and energy transport.

This study focuses on the evaluation of CUDA, a leading GPU programming framework, to accelerate PIC simulations in fusion research. We utilise the computational capabilities of HPC Vega, including CPU and GPU resources, and evaluate the impact of CUDA on the performance and portability of fusion-related PIC codes. Our analysis begins with a description of the challenges posed by fusion-related simulations and the importance of accurate plasma modelling. We then present our simplified PIC code tailored to the simulation of plasma sheaths in fusion devices and highlight its importance for understanding plasma-surface interactions.

By specifying initial conditions for the plasma density, temperature and electric field properties, we begin simulations aimed at investigating the behaviour of the plasma near the boundary. Key parameters such as particle motion and electric field calculations are analysed to evaluate the efficiency of CUDA in accelerating PIC simulations. Our results show the significant performance improvement of CUDA on NVIDIA GPUs and emphasise the potential of CUDA in accelerating fusion simulations. We also observe promising results in terms of code portability, indicating the adaptability of CUDA on different GPU architectures.

By accelerating fusion simulations with CUDA, researchers can accelerate progress towards practical fusion energy solutions. The insights gained from CUDA-based simulations provide valuable information for optimising fusion devices and overcoming important challenges in plasma physics.

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Exploring Parallelization Strategies for Monte Carlo Ray Tracing in Synthetic Diagnostics Analysis of Fusion Plasmas

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Monte Carlo Ray Tracing (MCRT) is a powerful method for calculating plasma radiative power deposition to plasma-facing surfaces and assessing the optical response of diagnostic systems in magnetically confined nuclear

devices. Its main advantage over other numerical methods is the ease of parallelization and the ability to qualitatively model surface reflections of rays. Through the use of synthetic diagnostics, i.e., considering radiation profiles simulated with plasma physics codes such as JINTRAC [1], SOLPS [2], and radiation from metallic surfaces, the MCRT method can be used to determine the synthetic signal on IR cameras. Large meshes describing detailed geometry of the first wall and dense plasma radiative profiles require intensive computational resources. A parallelization strategy needs to be developed to reduce ray-tracing calculations. In this paper, a new parallel MCRT study that assesses the synthetic signal of an IR camera in a nuclear reactor will be presented. The study takes as input the steady-state temperatures on the first wall panels of the ITER tokamak. These temperatures represent the thermal response of the first wall to plasma power deposition, assessed with field-line tracing techniques [3] and calculated using the ELMER FEM package [4]. The MCRT study then takes into account spatially distributed plasma sources and first wall temperatures, propagates rays through the computational domain, and assesses the power arriving at the camera sensor. Special attention is given to the parallelization of the MRCT method and its deployment to appropriate HPC infrastructure. The parallelization algorithm of MCRT is presented, as well as the scalability of the used MCRT code on appropriate HPC infrastructure without compromising the quality of the results.

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Nuclear fusion and plasma technologies

Development W-Cu composite for divertor applications

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According to the current design, the diverter consists of a serial array of rectangular units, tungsten monoblocks, connected to a copper alloy (CuCrZr) cooling tube running through the central region of the monoblocks. Tungsten serves as a functional plasma-facing armour material, whereas the copper alloy tube acts as a structural heat sink to remove the heat from the first wall. At high heat loads expected during the operation of the reactor (≥ 15 MW/m2), the temperature of the armour material exceeds the recrystallisation temperature of W, resulting in the formation of plastic low cycle fatigue (LCF) cracks. Furthermore, thermal stress accumulation at the W-Cu interface combined with neutron embrittlement of the Cu phase might result in component failure. The formation of a W-Cu composite and a gradual transition between W armour and Cu-based heat sink was proposed to lower the surface temperature and decrease the stress accumulation at the interface region, reducing the risk of crack formation and extending the lifetime of the diverter component.

W lattice structures based on triply periodic minimum surfaces (TPMS) were designed and manufactured using a laser-based powder bed fusion (LPBF) process to investigate the effect of internal structure and composition on the properties of the W-Cu composite material. As produced W lattice structures were infiltrated with molten Cu to realise interpenetrating W-Cu composites. Two types of unit cells were used, namely walled and skeletal diamond TPMS, with variations in porosity ranging from 10% to 90 %. A comparison between the designed and manufactured models was conducted, focusing on wall thickness and skeletal density, to asses the manufacturing accuracy. The manufactured structures showed differences in wall thickness compared to the design specifications, with deviations observed in all samples. This discrepancy revealed a proportional relation, where an increase in wall thickness correlated with a decrease in the specified designed thickness. Similarly, it was discovered that the porosity of the final products was lower than the designated porosity. Following infiltration with molten Cu, samples were analysed in terms of density and microstructure. Compressive strength testing was performed to investigate the effect of unit cell type (walled, skeletal) on the mechanical performance of the composite material. Experimental results were compared to the finite element model (FEM) used to predict the mechanical behaviour and determine the regions of stress concentration.

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Nuclear fusion and plasma technologies

DeHydraAC: design and current status

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We are building a new experimental beamline at the Micro Analytical Centre at Jožef Stefan Institute in Ljubljana, Slovenia. It will be installed at +20° exit from the switching magnet currently coupled with a 2.0 MV tandem accelerator. The beamline will be equipped with pair of slits, for the production of ion beams with low beam divergence. Also, electrostatic steerers will be incorporated for large-area sample irradiation. As the quality of produced ion beams is important for the end experiment, special care is given to beam diagnostics. The beamline is connected to the UHV experimental chamber named DeHydraAC (Defects and Hydrogen Analysis Chamber). The DeHydrAC is built on knowledge obtained in the currently operated INSIBA (IN-Situ Ion Beam Analysis) chamber [1]. Both are designed to study the dynamic and static properties of materials, with emphasis on the study of hydrogen isotope interactions with the materials. Compared to the old INSIBA the DeHydrAC is designed to operate at a vacuum of less than 10^{-9} mbar, which is important for studying hydrogen surface interactions with materials and for long ion beam irradiations. At this pressure, we reduce the surface contamination of the samples. To this end, we also installed the Auger spectrometer at the DeHydrAC. The combination of DeHydrACand the electrostatic beam scanner will allow ion beam irradiation of samples over large areas, mainly to produce neutron-like damage via a so-called self-bombardment process or to implant elements. A big design difference from old INSIBA is the addition of a 6-axes goniometer. The goniometer is built with a sample holder which allows us to study the samples in the temperature range from roughly 200 K up to 1300 K. The 6-axes goniometer was selected, for performing the precise channelling measurements, which hopefully can provide answers at which sites in the crystal lattice the hydrogen is retained at certain conditions. As the new DeHydrAC beamline is being built from the end of 2023 the current status will be presented and possible future experiments will be discussed.

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Nuclear fusion and plasma technologies

Neutronics simulations for the lower port area of the volumetric fusion neutron source

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As part of EUROfusion's efforts to develop fusion into a viable energy source, an initial concept for a European 14 MeV volumetric neutron source (VNS), a medium-sized tokamak-based fusion machine, was developed to mainly test tritium breeding blanket technologies under reactor conditions relevant for EU DEMO.

One of the challenging aspects is designing and integrating all relevant systems into the lower port area. This includes the vacuum pumping system, the divertor with its associated piping and ensuring there is enough space for divertor removal during maintenance using remote handling tools. At the same time, adequate protection of the superconducting coils, vacuum vessel and other important systems is required with regard to nuclear loads.

Neutronics analyses were carried out to investigate the suitability of the initial design choices and to propose alternative solutions where appropriate. The main parameters of interest were the nuclear loads in superconducting magnets and the helium production in divertor cooling pipes – a parameter that is crucial for the re-weldability of the pipes after maintenance or replacement of the divertor.

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Two-Phase Analysis of Critical Regions in the Cooling Circuit of the DEMO Divertor Cassette Body

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In the framework of the EUROfusion project on DEMO divertor development, detailed thermal-hydraulic analyses are needed to support the design and evaluation of the thermal-hydraulic performance of the DEMO divertor cooling system. In the previous work of Di Maio et al. [1], a detailed analysis of the divertor assembly has been performed by using single-phase Computational Fluid Dynamics (CFD) simulations. In this analysis, some local regions of high coolant temperature exceeding the local saturation margin have been found in the coolant circuit of the divertor cassette body (CB). This indicates that boiling or even critical heat flux conditions can be established in these regions.

Therefore, in this study, a two-phase flow analysis of the CB cooling system will be carried out to investigate the effects of two-phase flow phenomena in these critical regions. The ANSYS CFX finite volume code will be used to

perform the simulations. First, a full model of the Divertor CB with a cooling system will be used to repeat the single-phase simulations and confirm the locations of critical regions. Then, a two-phase simulation will be performed on the full or partial CB model containing the critical regions. The impact of the two-phase flow on the CB temperature profile, heat transfer and on the pressure drop in the cooling circuit will be evaluated. The two-phase flow results with boiling model will be compared with single-phase simulations. The boiling model and its modifications will be assessed against benchmark experiments from the literature beforehand.

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Nuclear fusion and plasma technologies

Upscaling study of W-W2C composite: a promising material for the DEMO divertor

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Tungsten (W), the selected core material for the DEMO`s divertor, suffers from high-temperature ductile-to-brittle transition, recrystallization and grain growth [1]. To address these issues, a breakthrough method involves the in situ formation of W_2C particles within the W matrix through WC doping. This innovative approach proves highly effective in eliminating detrimental W oxides while simultaneously enhancing densification. In addition, the refined microstructure improved the mechanical properties at room and high temperatures. The W_2C phase is homogeneously distributed within the W matrix due to carbon diffusion from WC nanoparticles during the sintering process. Even in an extremely fast sintering process such as Field-Assisted Sintering Technology (FAST, 1900°C, 5 min), the added WC completely transforms to W_2C, resulting in a tailored W-W_2C composite [2]. Notably, this composite exhibits a ductile-to-brittle transition temperature (DBTT) between 200 and 400°C, exceptional thermal stability, robust thermal shock resistance, and thermal transport properties within the required parameters. These qualities position W-4WC composites as an ideal material for the DEMO divertor application, showcasing their potential to revolutionize the landscape of fusion materials for future energy systems.

The upscaling studies, which are in progress, have demonstrated that it is possible to produce larger pieces (currently the upscale was done from pieces with diameter of 40 mm to pieces with diameter of 80 mm, while the thickness is as required for the monoblock production) also by FAST sintering, therefore the (semi)industrialization of the process is feasible and reachable.

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Acknowledgement

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them. This project has received funding from the Slovenian Research Agency (P2-0087-2 and P2-0405-5). The authors would like to thank Mr Matej Kocen for performing FAST sintering of samples.

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Nuclear fusion and plasma technologies

Analyses of radiation streaming paths for fusion applications

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One significant challenge in the support of operation of tokamaks devices is the leakage of radiation along streaming paths. This can create substantial radiation fields at considerable distances from the torus, complicating radiation protection for personnel and potentially impacting the heating of crucial tokamak components like superconducting magnets. There have been limited measurements of neutron streaming in larger tokamaks, which have been compared to calculations to establish benchmark experiments.

Modelling radiation transport through narrow pathways presents difficulties due to the significant reduction in neutron and gamma flux from the plasma source to remote areas far from the torus. As a result, variance reduction is crucial for these calculations. One common method for Monte Carlo simulations is adjusting particle weights during transport. The ADVANTG code, used alongside the MCNP code, is a cutting-edge tool for generating weight windows.

The work for the improvement of the coupled ADVANTG/MCNP approach by analysing individual particle histories, is presented. This technique provides additional data on the significance of specific regions along the pathways, which standard tallies cannot achieve. It also enhances the understanding of key particle paths and aids in interpreting measurement results more accurately. The research utilizes a simplified model of a large tokamak with various openings and streaming paths.

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Tungsten Detritiation using MSO Technology

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In the fusion energy industry, tritium is used as a fuel element and is essential for the future operation of the first generation of fusion power plants. Tritium is very rare in nature and is currently recovered from the coolant of heavy water nuclear reactors. The recovery of tritium is a complex and costly process involving several steps, from the separation of tritium from heavy water to its purification and storage, which is severely limited by the low half-life of tritium. Tritium management and methods for its recovery and eventual recycling remain a challenging issue. Efficient tritium management is essential for the safe and economic operation of future fusion power plants. The operation of experimental fusion reactors has demonstrated the production of radioactive waste in the form of various tritium-contaminated materials. The management of tritium-contaminated materials has received considerable attention in research into the energetic use of thermonuclear fusion. A significant type of radioactive waste that will be generated during the lifetime of a fusion power plant is tritiated tungsten dust. In this paper, experiments have been carried out to determine the effectiveness of the Molten Salt Oxidation (MSO) process in recovering tritium from contaminated tungsten pellets simulating tritiated tungsten dust. Analysis of the output showed successful capture of tritium in the form of tritiated water and determined the efficiency of the process under defined conditions. A two-stage water condensation system was used to capture tritium. Tritium capture efficiencies range from 10 to 37 % with a median efficiency of 13.82 %. More than 90 % of the activity was captured in the first stage of the condensation system. This process has proven to be an effective way to treat and recycle tritium contaminated materials. From the data obtained, a possible approach to the treatment of the expected radioactive waste from fusion power plants is presented. The use of MSO technology for the purpose of tungsten detritiation appears appropriate and its effectiveness

has been experimentally demonstrated. The next steps include extensive testing in different operational settings to optimize the process for industrial applications.

Nuclear power plant operation and plant life management

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Nuclear power plant operation and plant life management

Assessment of Selected Long-Term Operation Improvements Relevant to the Pressurized Thermal Shock in PWR with Focus on Reactor Pressure Vessel Integrity

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The impact of nine different long-term operation (LTO) improvements relevant to the pressurized thermal shock (PTS) phenomena in a pressurized water reactor (PWR) was investigated. The main focus is on the reactor pressure vessel (RPV), which is the most vulnerable to the PTS and beltline weld. Six LTO improvements are connected with the change in parameters of the safety systems: a heating of water in high-pressure injection system (HPSI) tanks, a heating of water in accumulators, a heating of water in low-pressure injection system (LPSI) tanks, a decreasing HPSI pump head, a decreasing HPSI capacity and a decreasing the accumulator pressure. The remaining three LTO improvements are related to human factors, and they are each modelled as operator actions: a reduction of HPSI system flow by the operator, a different secondary-side cooldown rate and an isolation of accumulators.

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 plant studied in the analysis is based on a 1300 MW four-loop PWR German design.

The thermohydraulic (TH) impact of the LTO improvements was studied with the RELAP5/Mod3.3 computer code with the two-dimensional nodalization is applied to the RPV downcomer section to enable the modelling of the asymmetric cooldown of the RPV. Deterministic structural mechanics calculations were conducted using the FAVOR code. The fracture mechanics was studied using the CEA method and in-house developed tools.

This paper studies the impact of the proposed LTO improvement on stress intensity factors in specified points of the postulated cracks (through-clad cracks and under-clad cracks). Finally, maximum allowable adjusted reference temperatures are calculated for studied cracks, which were calculated using the ASME curve and the Tangent approach.

This work is a part of the larger benchmark performed in Work Packages 2 and 3 of the project Advanced PTS Analysis for LTO (APAL), which is a Euratom-funded research and training programme. This paper presents selected results prepared by the Warsaw University of Technology.

1102

Nuclear power plant operation and plant life management

Assessment of the optimal electric power of the new nuclear power plant JEK2 for secure and stable operation and development of the electric power system of Slovenia

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The paper examines the electric power system connection feasibility of different single and double configurations of the new nuclear power plant in Krško (JEK2) with a range of total net electric power from 1.000 to 2.400 MW. This paper considers connection of JEK2 to the grid in 2040 and in relation to this, the crucial operating states of the electric power system are set, considering the current long-term energy forecasts and current regulatory frameworks of the electric power system operation. Stationary analyses examine the states with and without JEK2, evaluate the impact of JEK2 on the network and identify the needs to upgrade transmission system paths. The impact of JEK2 is also verified by short-circuit and dynamic analyses. The paper includes an analysis of future needs and mechanisms to ensure the frequency restoration reserves. Finally, economic evaluations of investments in the transmission network and the frequency restoration reserves are provided. The paper is concluded with key findings and proposals for further work, which are necessary for the reliable connection of JEK2 to the electric power system and the successful implementation of the project.

1103

Nuclear power plant operation and plant life management

Cable Aging Management Program and Development of Acceptance Criteria for Radiation Aged Field Assessment of Polymers in Krsko Nuclear Power Plant (NEK)

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One of additional requirements for plant life extension above 40 years of operation, additional Cable Aging Management Program (CAMP) has to be implemented in Krško NPP. Electrical cables represent an important component of the Nuclear Power Plant (NPP) nuclear island as well as of its balance of plant. Therefore, it is reasonable to monitor their aging process and expected reliability. There are, however, many electrical, thermal, mechanical, radiation and humidity stressors in localised adverse environment, which, through time, can influence the occurrence of insulation degradation. Assessment and evaluation of the condition of passive cable insulation is supposed to be carried out periodically, during the maintenance activities on active components. Accordingly, it is necessary to carefully select the proper efficient diagnostic methods for determining the aging grade and overall adequacy of the cable insulation. To avoid unplanned failures with loss of critical equipment important for safety, the insulation must be properly tested. Therefore electrical and non-electrical diagnostic methods are used to ensure proper assessment of cable insulation deterioration. This paper reports on experience with application of different diagnostic methods boosting tendency to improve monitoring of the condition of the cable insulation in the Krško NPP. The environmental monitoring, diagnostic testing and measurements on cables recommended to be done in nuclear power plant respecting a good practice of asset maintenance management is discussed. Standard tests exist for the analysis of cable properties, such as measurements of insulation resistance, dielectric losses, partial discharges, time-domain reflectometry, and frequency-domain reflectometry, each with its weaknesses and benefits. In addition to electrical measurements, mechanical tests are also conducted, such as the intender modulus based on the tensile test and measurements of the cable diameter. Electrical and mechanical changes are caused by chemical changes in the material resulting from molecule decay and the extraction of molecules due to these changes. For these purposes, techniques such as differential scanning calorimetry (DSC), Fourier-transform infrared spectroscopy (FTIR), and X-ray fluorescence spectrometry (XRF) are employed. This paper reports on development of field diagnostic method for assessment of simultaneous aging effects of radiation and temperature aged cable polymer insulation using mechanical properties such as Indenter Modulus (IM). Experiments were prepared with samples of Nuclear Safety Related qualified cables from different manufacturers (Boston Insulated Wire and Rockbestos) and different vintages (1976, 1979 and 2011). Nuclear safety related cables were manufactured and qualified with most frequently used manufacturers and polymer materials such as Ethylene propylene rubber (EPR) and crosslinked polyethylene (XLPE) all with Chlorosulphonated (CSPE) Jacket. Samples were taken on 4 different aging stages with separated gamma and neutron irradiation at 500 kGy, 1000 kGy, 1500 kGy and 2000 kGy. Radiation aging was conducted at working Triga temperature up to 54°C leading to basis for simultaneous adverse environment

ageing effects of high radiation and moderate temperature that was recognized in previous researches as potential negative effects for specific materials. Radiation was also separated on neutron and gamma effects and for each sample indenter modulus value was tested in all different aging steps with intention to get acceptance criteria for remaining life of exposed polymers. Results of diagnostic testing method showed an evident change of properties in polymers used in jacket as leading indicator for cable insulation. As presented results and methods could be used in nuclear power plant to determine remaining life of polymers exposed to degradation due to simultaneous effects of temperature and radiation.

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Nuclear power plant operation and plant life management

Indicative pre-investment economic analysis of the project JEK2

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The construction of the new nuclear power plant in Krško (JEK2) is a key strategic development project of the GEN Group. Studies carried out so far have shown that JEK2 is a feasible project for the future security of Slovenia's supply of domestically produced electricity without greenhouse gas emissions, and that it adequately responds to the key challenges of the energy trilemma of growing demand, the phasing out of fossil fuel-based energy and the limitations of storing electricity generated from variable solar and wind power. The purpose of this contribution is to present the JEK2 project from an economic point of view based on currently known and available data, obtained through a process of expert dialogue with suppliers, and considering comparable data from the existing Krško Nuclear Power Plant.

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Nuclear power plant operation and plant life management

SI-53 direct cause analysis report

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A section of piping from safety injection line SI-53 from NEK was destructively examined to investigate the cracking degradation associated with a leak in the pipe. Extensive failure analysis was performed for two main cracks.

1106
Nuclear power plant operation and plant life management

The benefits of standardization & increasing of manufacturing capacities to serve the logic of the European nuclear fleet

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Offering the highest standards of safety, EPR family benefits from decades of Framatome's experience in designing, manufacturing and delivering components for nuclear pressurized water reactors worldwide. Framatome duty is to deliver NSSS components capitalizing on lessons learned from EPR projects currently in operation. As reliable manufacturer, Framatome has a proven and recent track record of projects delivered in Europe and world wide. Being able to address European needs for decarbonation through EPR nuclear new build projects as well as being able to supply replacement components for long term operations are key decision factors. To avoid uncertainties in managing such important projects, Framatome has standardized components manufacturing and increased its manufacturing capabilities in order to be able to serve the logic of European nuclear fleet with same standard components & solutions. In Slovenia, Framatome, with support of its partners, is ready to deliver the promise.

1107

Nuclear power plant operation and plant life management

Review and Analysis of Organizational Charts and Personnel Management for the Nuclear Power Plant Krško 2

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The article presents the organizational charts and personnel management of various nuclear power plants worldwide with the aim to establish appropriate the project organizational chart for the Nuclear Power Plant Krško 2 (hereinafter: JEK2).

The article introduces the theory of companies' organizational charts, highlighting the general concept, types, importance, and usefulness, while presenting personnel employment needs for the nuclear power plant construction period. Examples of nuclear power plant organizational charts from Turkey, Hungary, Brazil, South Korea, Slovakia, and Slovenia are presented to evaluate their applicability for JEK2. At the same time, the focus is put on other Slovenian construction projects and their organizational schemes to expose the national peculiarities in building such projects.

In the last chapter, we present, discuss, and evaluate the already prepared organizational scheme for JEK2 and provide suggestions for improvements based on examples of good practices from abroad.

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Nuclear power plant operation and plant life management

Replacement of a part of SI pipeline in NPP Krško

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In October last year, a leak occurred in the SI pipeline, leading to an unplanned shutdown and unplanned outage. In a very short period, NEK (NPP Krško), with the help of Westinghouse and its company PCI, along with Slovenian companies Numip and Q Techna, executed the replacement of a section of the SI pipeline. This article will highlight the key moments, correct decisions, sound engineering approach, appropriate organization, and best practices that led to a successful completion and a record-breaking quick restart.

1109
Nuclear power plant operation and plant life management

Calculating Uncertainties of Environmental Qualification Instrument Channels in NPP Krško (NEK)

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The reliable and safe operation of nuclear power plants (NPPs) heavily depends on the performance of control and measuring equipment. When designing individual systems and their associated components, it is necessary to consider the actual capabilities (accuracy or errors) and limitations of the I&C components and/or entire I&C loops. Calculating instrument channel uncertainties for various instrument loops is therefore essential.

While uncertainty calculations for I&C channels involved in reactor protection or safety limits are typically included in the original license or project documentation, and updated accordingly with changes, other I&C channels, primarily used for mitigating consequences of abnormal operating scenarios or accidents not initially anticipated, may lack full uncertainty assessments.

This article presents a practical case study of instrument channel uncertainty calculation for various environmentally qualified (EQ) instrument channels in NPP Krško (NEK) that have not yet undergone such analysis. The calculation aims to identify the increase in uncertainty for EQ instrument channels due to demanding environments.

The article begins with a general overview of the Equipment Environmental Qualification in NEK, followed by a brief presentation of instrument channel uncertainties, outlining relevant rules, practices, and standards guiding instrument channel uncertainty calculations. The core part of the article presents a case study calculating uncertainties for EQ instrument channels in NEK. It discusses the approach, methodology, calculation process, results, and their significance for NEK's.

1110
Nuclear power plant operation and plant life management

Metroscope: software for monitoring and diagnostics of industrial assets

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Metroscope, a subsidiary of the EDF Group, specializes in monitoring and diagnostics of industrial assets through its advanced Digital Twin technology. This technology allows for the identification of over 150 faults monthly with a 90% reliability rate, significantly enhancing process efficiency. Metroscope's solutions are applied across all 56 EDF nuclear power plants in France, yielding annual savings exceeding 34 million euros. The company also serves over 77 plants globally, both nuclear and conventional, with a user base exceeding 450 individuals, and partners with industry leaders Framatome and Microsoft.

During the conference, Metroscope will introduce its identity, value proposition, technology, and notable references. The focus will be on their innovative approach to automated power plant diagnosis using Artificial Intelligence, a technology derived from EDF R&D and recognized with the 2024 WNE Innovation award. Metroscope's mission centers on eliminating unplanned outages and energy losses through AI.

The presentation will delve into Metroscope's strategies for tackling challenges in nuclear power utilities, emphasizing reliability, availability, and performance. As a software editor, Metroscope will showcase its

capabilities in root cause analysis and early fault detection, supported by case studies on condition-based maintenance, performance management, and early-stage detection.

Regulatory issues and legislation

1201

Regulatory issues and legislation

Licensing procedure for a new nuclear power plant JEK2

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The licensing for the JEK2 project in Slovenia is a complex process performed in several phases and it involves the ministry competent for the spatial planning, and the ministry competent for the environment and energy, as well as other administrations, which are included in the different licensing processes. The Slovenian Nuclear Safety Administration (SNSA) is the responsible authority for radiation and nuclear safety of the project.

The paper will present the legislative framework in the licensing process for the new nuclear power plant JEK2. For this purpose, the SNSA applies the Ionising radiation protection and nuclear safety act and rules, developed from the act. More detailed are the practical guidelines of the SNSA, among them also the new Guidance for the Siting of Nuclear Installations, which is in preparation and is based upon the relevant IAEA safety standard.

The licensing process for the JEK2 project is divided into four phases. The first phase involves the siting of the new nuclear installation, which is performed through the preparation of the National Spatial Plan (NSP) for the facility. This process is managed by the ministry responsible for spatial planning and includes a number of spatial planning authorities with their experts, which provide guidelines for the siting of the new nuclear facility. The role of the SNSA is the preparation of guidelines for the siting of JEK2 in order to issue an opinion on the documentation for NSP at the end of the process. The SNSA is involved also in the process of comprehensive assessment of environmental impact. The second phase in the JEK2 licensing project is obtaining a building permit where the process is managed by the ministry responsible for building. SNSA participation in the building permit process is very important and starts with preparation of the design conditions for the JEK2 design and construction. The main role of the SNSA is review and approval of the preliminary safety analysis report (SAR) for the JEK2. The safety analysis report is an important and comprehensive document containing key information on the nuclear installation, its operating limits and conditions, its impact on the environment, a description of the project, analyses of possible accidents and the measures needed to prevent or mitigate the impact to the environment, the public and the workers. In the process of construction, the SNSA approves the program of the commissioning for the facility. In parallel, the environmental impact assessment is performed, where the SNSA has its role in preparing the requirements for the monitoring of the JEK2 radiological impact to the environment. The last two phases are licensing of trial operation and the process for the operation license of the facility that is issued by the SNSA.

The paper will discuss the different phases in the licensing process together with related acts and regulations. The contents will focus mostly on the first two phases of the licensing, namely the siting and the construction licenses. The role of the SNSA in the licensing of the project JEK2 will be presented.

1202
Regulatory issues and legislation

Map Determination of Code & Standard Needs to be Covered for Innovative Nuclear Reactors

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In 2023, the nuclear industry was clearly recognized as having a key role to play in turning the tide on global warming. For the first time, nuclear energy was explicitly mentioned as a solution in the final COP28 declaration and 29 nations have appealed for a concerted effort to triple the nuclear capacity by 2050. In the European Union, for historical reasons, the nuclear reactors in operation today were designed and built using either the US ASME/ASCE Codes, European codes (AFCEN for France, KTA for Germany), Russian codes (PNAE G-7), or in some cases a mixture of codes adapted at a national level. This led to a European nuclear reactors fleet built with a "patchwork" of reference codes, causing in some cases higher costs for spare parts and more complex follow-up of activities. As of 2017 and mentioned by the EC Directorate-General “Energy” in its Nuclear Illustrative Programme (PINC), this situation is hampering the competitiveness of the nuclear industry and therefore the deployment of new capacities.

Nuclear energy provides provide low-carbon and plannable electricity as well as industrial heat and energy vectors (notably hydrogen). New light-water reactors of Generation III are needed as well as more innovative Generation IV reactors, and another trend is that small modular reactors (SMR) will take a considerable share. This need for harmonisation is essential for SMR projects insofar as the business model for this type of reactor relies on modularization and series production in addition to design simplification.

The HARMONISE project is a research and development initiative supported by Horizon Europe which aims to accelerate the implementation of innovative nuclear technologies, identify the gaps that need to be filled to favour the licensing of those technologies under development and which could contribute to the European energy mix in the short/medium term.

One of the WPs in this project deals with the subject from the point of view of codes and standards. Innovative reactor projects call for new technologies based on different and often challenging operating conditions from innovating reactors and require special component designs. These three reasons give rise to new needs in terms of codes and standards that need to be addressed and implemented in a very short timeframe. The work carried out as part of this Project consists, first and foremost, of identifying these needs. To this end, initial data were collected from project partners involved in innovative reactor projects such as DEMO and ALFRED, and a questionnaire was sent to reactor developers, in particular to the many start-ups that have been created in recent years.

These needs are processed to draw up a questionnaire that will be sent to the Standard Development Organisations (SDOs). For the last two categories, the SDOs are expected to provide methodologies. In parallel, the questionnaire will also be sent to other industrial sectors such as aerospace, automotive, rail, etc... In this case, the interest is in seeing how new technologies (essentially digital and manufacturing technologies) have been taken into account and in investigating the possibility to transfer to nuclear field code and standard rules successfully developed.

At the end, a map will be formulated describing the evolution of the codes and standards to fill the gaps and address the development needs identified and implementing the solution options identified. The emphasis will be placed on identifying potential improvements in the harmonisation of codes and standards that could be adopted when the proposed innovations are implemented.

1203
Regulatory issues and legislation

A Quick Outline of the National Approaches and Milestones vis-à-vis IAEA INFCIRC/908, INFCIRC/910 and INFCIRC/918

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The Slovenian stakeholders have rich national and international experiences with different subsets of safe and secure use of radioactive sources and nuclear material, including fissile material. The "Nuclear Law" gives the formal platform for nuclear safety, radiation safety and nuclear security (physical protection as a narrower term) too. Internationally, the International Atomic Energy Agency (IAEA) provides for extensive co-operation embracing all aspects of peaceful use of nuclear energy. Other international organisations, fora, export control regimes, and associations knead and chisel those commitments and ideas which may be transferred to further pledges and obligations. One of the off springs of the Nuclear Security Summits were various pledges, commitments and initiatives that have brought like-minded countries together and shed lights around a few underrated subsets to be underpinned so as to further strengthen nuclear security and some interfaces. They were channelled through IAEA to be later-on published through its avenues, namely INFCIRCs (Information Circulars).

In the previous decade, the Slovenian Nuclear Safety Administration (SNSA) were in due contacts with the representatives from the Ministry of Foreign Affairs and to certain degree also with the Ministry of the Interior. Three international endeavours were assessed to be of particular importance, to be "hand-picked" and recognised through the official channels. Within the interval of less than three years, Slovenia joint the following international initiatives. IAEA INFCIRC/908 encompasses mitigating insider threats (in the nuclear sphere). INFCIRC/910 focuses on strengthening the security of high activity sealed radioactive sources. Lastly, INFCIRC/918 addresses countering nuclear smuggling. There are other initiatives that gained the impetus after 2016 and connect different countries (and organisations) around the globe to enhance specific subsets, e.g. transport of nuclear material or regarding further use of highly enriched uranium.

The article collates and evaluates a myriad of sometimes "stochastic" (or *ad hoc* actions) and a set of traditional, well-established and matured tasks, tools and approaches to maintain some national networks in Slovenia, nurture tailor-made outreach activities and respond to those international actors looking for specific types of co-operation or engagements.

SNSA has been fairly active domestically as well as internationally, having in mind its limited resources in this vein. It is of utmost importance to interweave resilience and sustainability in nuclear security-related activities which are multi-pronged and connected in a number of cases with other, adjacent areas – so as to amalgamate 3S approach efficiently (safety – security – safeguards). This year's conference in Vienna, namely International Conference on Nuclear Security: Shaping the Future is one of the fulcrums and stepping-stones for a string of activities to shape the very next endeavours. Both the top-down as well as bottom-up approaches are necessary to "mind the gaps" and continuously advance. And yes, it is always possible to further enhance dialogues with different national stakeholders (looking for synergies), promote IAEA's documents and mechanisms and last but not least, with due care, professionalism and exactness: shape safe and secure use of radioactive sources and nuclear material, spanning from transport to outreach to scrap recyclers, from adjusting administrative burdens to reply to various international queries and questionnaires.

1204
Regulatory issues and legislation

What we learned from the Krško NPP Spent Fuel Dry Storage Project - the Regulatory Aspect

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Following the Fukushima Daiichi accident, the Krško Nuclear Power Plant (NEK) undertook a significant safety upgrade project, including a new spent fuel dry storage facility (SFDS). This paper describes the licensing process which was done by the Slovenian Nuclear Safety Administration (SNSA) for advanced spent fuel storage solution.

The SFDS design incorporates the latest safety requirements, standards and the best practices for on-site spent nuclear fuel storage. To ensure the highest safety standards, the project incorporated a customized design in compliance with the Krško NPP's requirements with the latest design extension conditions (DEC) concept.

Licensing process proved to be demanding. Public participation, encompassing neighboring countries and local residents, ensured transparency and fostered trust in the project.

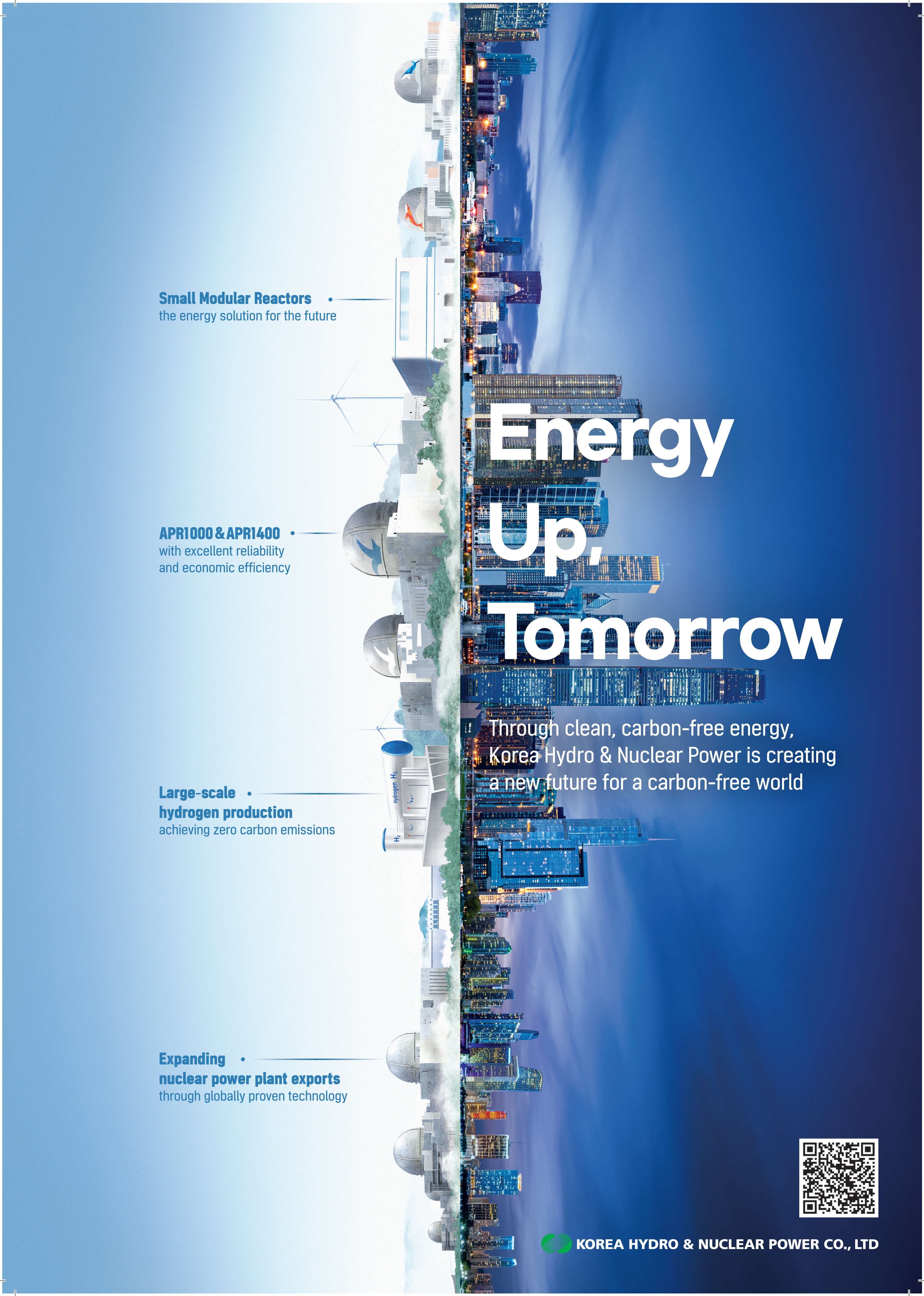
The SNSA's licensing process was a multi-step process with emphasis on several key aspects such as the stringent post-Fukushima safety requirements, the comprehensive safety assessments, the rigorous oversight and the in-depth inspections. The project adhered to new safety regulations implemented after the Fukushima Daiichi accident. Independent expert evaluations focused on seismic risks and radiological loads which were crucial for the licencing approval. A strict project design basis conditions for construction, operation and conditions for fuel transfer procedures were set by the SNSA and other participants. The procedures also include the personnel training, the equipment certification and the dry run tests.

During the construction and the first fuel transfer from spent fuel pool to the SFDS, the SNSA conducted thorough inspections to verify that construction is in accordance with the design bases conditions.

The result of successful licensing process and comprehensive inspections was the safe first transfer campaign of spent fuel from spent fuel pool to the SFDS in 2023.

The experience gained serves as a valuable example for future nuclear facility licensing efforts, including potentially more complex projects like the proposed new NPP JEK2.

The content of the paper will focus mostly on the SNSA experiences regarding the licensing process of the Krško NPP spent fuel dry storage project.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