NENE

International Conference
Nuclear Energy for New Europe

11th -14th September 2023
Portorož, Slovenia

# Book of Abstracts

**Nuclear. The surest path to a brighter future.**

32nd International Conference
Nuclear Energy for New Europe

www.djs.si/nene2023

# Sponsors

## Platinum sponsors

## Gold sponsor

## Silver sponsors

## Bronze sponsors

## Sponsors

# Book of Abstracts

## Organisers:

Nuclear Society of Slovenia
Društvo jedrskih strokovnjakov Slovenije

And

GEN GROUP

In collaboration with: ENS EUROPEAN NUCLEAR SOCIETY

## Contact


**Društvo jedrskih strokovnjakov Slovenije**
**NENE2023**

Jamova cesta 39
SI-1000 Ljubljana
Slovenia

**E :** nene2023@djs.si
**W**: www.djs.si/nene2023

## Publisher

Društvo jedrskih strokovnjakov Slovenije
Jamova cesta 39
SI-1000 Ljubljana
Slovenia

e-mail: djs@djs.si
www: www.djs.si

## Editors

Tomaž Žagar
Melita Lenošek Kavčič

## Cover Photo

Shutterstock

## Edition

400 copies

## Printed by

Tiskarna Jordan

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

621.039(4)(082)

INTERNATIONAL Conference Nuclear Energy for New Europe (32 ; 2023 ; Portorož)
Book of abstracts : [32nd International Conference Nuclear Energy for New Europe NENE - 2023, Portorož, Slovenia, September 11-14] / [editors Tomaž Žagar, Melita Lenošek Kavčič ; organizers Nuclear Society of Slovenia]. - Ljubljana: Društvo jedrskih strokovnjakov Slovenije, 2023

ISBN 978-961-6207-55-3
COBISS.SI-ID 161694211

# Committees

## Program Committee

**Chair: Tomaž Žagar, Slovenia**

Didier Banner, France
Robert Bergant, Slovenia
Leon Cizelj, Slovenia
Michèle Coeck, Belgium
Marko Čepin, Slovenia
Štefan Čerba, Slovak Republic
Christophe Destouches, France
Antonella Di Trapani, OECD NEA
Francisco Javier Elorza, Spain
Bruno Glaser, Slovenia
Davor Grgić, Croatia
Pavlin Petkov Groudev, Bulgaria
Tomaž Gyergyek, Slovenia
Igor Jenčič, Slovenia
Gonzalo Jiménez, Spain
Romana Jordan, Slovenia
Ivo Kljenak, Slovenia
Ivan Kodeli, United Kingdom
Boštjan Končar, Slovenia
Božidar Krajnc, Slovenia
Eileen Langegger, Austria
Gabriel Lazaro Pavel, ENEN
Matjaž Leskovar, Slovenia
Rosa Lo Frano, Italy
Abdallah Lyoussi, France
Andreja Peršič, Slovenia
Gorazd Pfeifer, Slovenia
Igor Pioro, Canada
Igor Simonovski, EC DG JRC
Radek Škoda, Czech Republic
Luka Snoj, Slovenia
Luka Štrubelj, Slovenia
Iztok Tiselj, Slovenia
Andrej Trkov, Slovenia
Krešimir Trontl, Croatia
Sean Tyson, United States of America
Timothy E. Valentine, United States of America
Sandi Viršek, Slovenia
Branislav Vrban, Slovak Republic
Franck Wastin, The Netherlands
Nadja Železnik, Slovenia

## Advisory committee

Gerard Cognet, France
Nikola Čavlina, Croatia
Francesco D'Auria, Italy
Michel Giot, Belgium
Yassin A. Hassan, United States of America
John E. Kelly, United States of America
Danijel Levičar, Slovenia
Borut Mavko, Slovenia
Saša Medaković, Croatia
Irena Mele, Slovenia
Martin Novšak, Slovenia
Stanislav Rožman, Slovenia
Igor Sirc, Slovenia
Joerg Starflinger, Germany
Andrej Stritar, Slovenia

## Young Author Award Committee

**Chair: Andrej Trkov, Slovenia**

Didier Banner, France
Aleš Buršič, Slovenia
Davor Grgić, Croatia
Rosa Lo Frano, Italy
Luka Štrubelj, Slovenia

## Best Poster Award Committee

**Chair: Ivo Kljenak, Slovenia**

Roberta Cirillo, ENEN
Christophe Destouches, France
Jure Kos, Slovenia
Tanja Urek, Slovenia

## Organizing Committee

**Chair: Melita Lenošek Kavčič, Slovenia**

Nataša Dular, Slovenia
Pia Fackovič Volčanjk, Slovenia
Mateja Južnik, Slovenia
Jure Kos, Slovenia
Jan Lokar, Slovenia
Mojca Marija Petrovčič, Slovenia
Tanja Urek, Slovenia
Boris Vovčko, Slovenia
Kaja Zupančič, Slovenia
Rok Županc, Slovenia

## 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
- Nuclear Energy for New Europe 2010, Portorož, Slovenia, September 2010
- Nuclear Energy for New Europe 2011, Bovec, Slovenia, September 2011
- Nuclear Energy for New Europe 2012, Ljubljana, Slovenia, September 2012
- Nuclear Energy for New Europe 2013, Bled, Slovenia, September 2013
- Nuclear Energy for New Europe 2014, Portorož, Slovenia, September 2014
- Nuclear Energy for New Europe 2015, Portorož, Slovenia, September 2015
- Nuclear Energy for New Europe 2016, Portorož, Slovenia, September 2016
- 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

# Welcome

Dear participants,

Welcome to the 32nd conference Nuclear Energy for New Europe. Traditionally organised by the Nuclear Society of Slovenia, this conference is an international meeting of nuclear experts dealing with different aspects of nuclear energy. The primary objective of the meeting is to foster international cooperation amongst professionals from nuclear utilities, nuclear research organisations, educational institutions, industrial companies, and regulatory bodies.

Nuclear. The surest path to a brighter future. With this conference slogan, we emphasise the importance of nuclear energy as a key clean energy source, enabling countries to achieve ambitious international and national climate and energy goals effectively and efficiently. But not only climate and energy goals. Nuclear energy is furthering all seventeen United Nations sustainable development goals, including healthcare, food security, access to clean water, industry, and economic growth.

The conference program begins with lectures on nuclear energy's role in national energy and decarbonisation strategies of France and the United Kingdom, two countries with very serious climate and energy independence goals. For nuclear energy to achieve its full potential globally, nuclear knowledge and expertise growth will be needed in several parallel development pillars. The conference program features distinguished speakers who will delve into key topics addressing these development pillars: long-term operation of existing large reactors, activities related to planning and construction of new reactors, and development of future generation reactors for even more versatile applications. A systematic approach to human capacity building across all those pillars will be required to reach sustainable growth and stability in nuclear organisations.

Additionally, contributed papers will offer valuable insights into the latest scientific research and technological advancements in energy and decarbonisation policies, nuclear power plant operations, outage management, lifetime extensions, nuclear regulation, reactor physic and research reactors, thermal hydraulics, probabilistic safety analyses, nuclear materials, fuel cycle and radioactive waste, nuclear education, training, and nuclear leadership. The conference will emphasise and discuss the opportunities and challenges associated with all aspects of peaceful uses of nuclear energy and nuclear power generation.

All these topics are timely and relevant for Slovenia, with the recent final approval of a lifetime extension for NPP Krško, with active planning of new reactors in Slovenia and intensive development of future generations of nuclear experts. The record-breaking number of participants in the young authors contest at this conference also shows how the young generation of experts believes in nuclear power as a secure path to a clean, bright energy future.

We look forward to your attendance at the conference.

**Melita Lenošek Kavčič**
Organizing committee chair

**Dr. Tomaž Žagar**
Program committee chair
Nuclear Society of Slovenia President

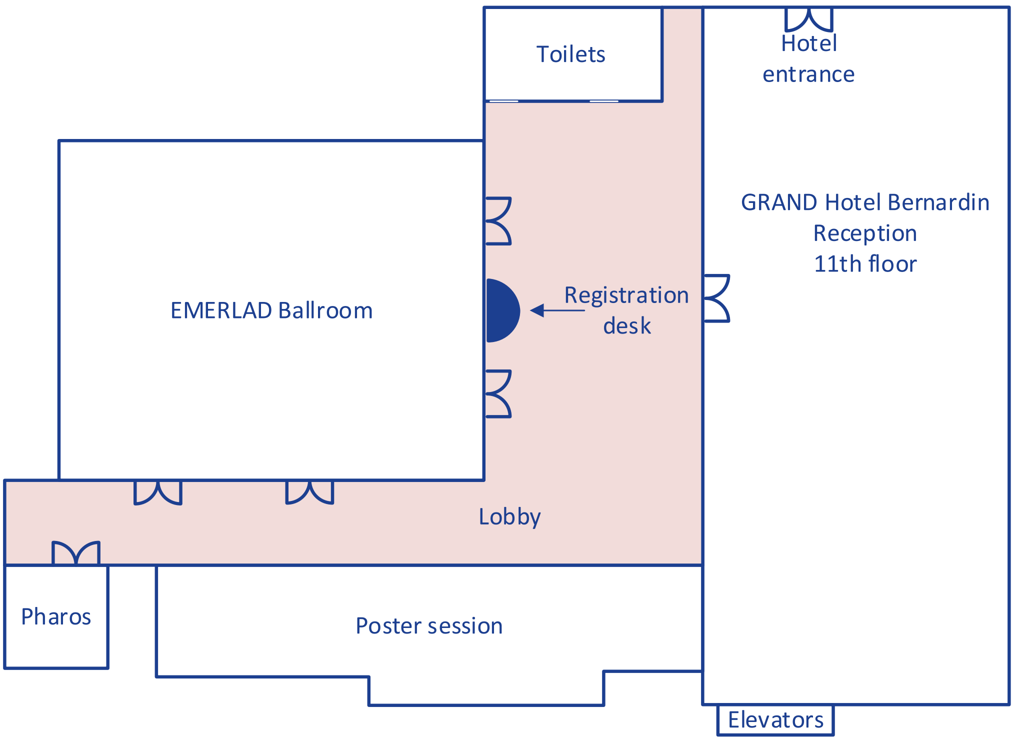
# Conference Location and Time

This conference will be hosted at the Grand Hotel Bernardin in Portorož, Slovenia. Renowned as Slovenia's premier convention hotel, GH Bernardin offers a remarkable venue nestled near the serene coastline, fostering an environment that is both conducive to work and inspiring in nature.

**St. Bernardin Resort**
**Grand Hotel Bernardin**
Obala 2
6230 Portotož

From: Monday, September 11, at 15:00
To: Thursday, September 14, at 12:30


Lectures will be held in the Emerald Ballroom and poster sessions in the adjacent Mediteranea and Adria Convention Halls on the **11th floor of Grand Hotel Bernardin**.



# Guidelines to Authors

## Oral Presentations

**Oral** presentations will be limited to 15 minutes. The authors are kindly asked to present their papers in **10 minutes** to allow 5 minutes for discussion. The participants are advised to prepare their oral presentations in **Microsoft PowerPoint format**. A laptop PC with Microsoft PowerPoint will be available. If additional equipment is required, please contact the organizers.

## Poster Presentations

**Posters** should be prepared to fit within **95 cm (width) x 110 cm (height)**. Posters should be posted and removed from the panels by the authors. Authors are kindly requested to post their presentations by 8:30 on Tuesday, September 12 and remove them no later than 12:00 on Thursday, September 14. Special poster sessions are scheduled on Tuesday, September 12, 2023, from 10:20 to 11:00, on Wednesday, September 13, 2023, from 10:20 to 11:00, and on Thursday, September 14, 2023, from 10:20 to 11:00.

# Publications

## Book of Abstracts

Each participant will receive the Book of abstracts in printed form. The Book of abstracts will also be available in the electronic form at the conference web page.

## Proceedings

Proceedings containing full length peer reviewed papers presented at the conference will be published in electronic form after the conference and mailed to the participants.

# Awards

## Young Author Award Contest

An award will be presented for the best paper prepared by a first author aged **no more than 32 years**. The number of co-authors should be limited to a maximum of two additional co-authors (advisers or mentors). The Young Author 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. The posters will be evaluated according to clarity of objectives, results and conclusions, scientific relevance, as well as according to aesthetics and attractiveness.

# Registration

## Registration Desk Opening Hours:

| | |
|---|---|
| Monday, September 11: | 13:00 to 19:00 |
| Tuesday, September 12: | 8:00 to 18:00 |
| Wednesday, September 13: | 8:00 to 18:00 |
| Thursday, September 14: | 8:00 to 12:00 |

# Social Activities

All participants are cordially invited to participate in all social activities.

## Conference Lunches **Tuesday, Wednesday and Thursday**

Lunch on Tuesday, Wednesday and Thursday is included in the registration fee and will be served from 12:30 to 13:45 at Sunset Restaurant on the 10th floor of the Grand Hotel Bernardin.

## Welcome Reception **Monday, September 11**

The Welcome Reception with welcome drink will be held on Monday at 19:30 on the Grand Garden Covered Terrace on 11th floor of the Grand Hotel Bernardin.

## Social Event **Tuesday, September 12**

A unique opportunity for socializing and bonding will be a walking tour to Piran, during which participants will get to know the many peculiarities of this old town. The meeting point is in front of the entrance to GH Bernardin from the seaside (ground floor), Tuesday, 12th September at 18:30.



Photos: Shutterstock

## Conference Dinner **Wednesday, September 13**

The conference dinner will be held on Wednesday at 19:30 at Sunset Restaurant on the 10th floor of the Grand Hotel Bernardin. The Young Authors Award and the Best Poster Award will be presented during the dinner.

# Special Events

## Group Photo

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

## Young Author Contest Award Ceremony

The award ceremony will be held on Wednesday, September 13, during the conference dinner. The chair will present the Young Author Contest Committee decision and give the award to maximum of two authors.

## Best poster Award Ceremony

The award ceremony will be held on Wednesday, September 13, during the conference dinner. The chair will present the Best Poster Award Committee decision and give the award to maximum of two authors.

## Conference Timetable

Lectures will be held in the **Emerald Ballroom** and poster sessions in the **Mediteranea** and **Adria Convention Hall** on the 11th floor of the Grand Hotel Bernardin.

| Monday | Tuesday | Wednesday | Thursday |
|---|---|---|---|
| | Registration 8:00-8:30 | Registration 8:00-8:30 | Registration 8:00-8:30 |
| | Invited (Nuclear New Builds) 08:30-09:00 | Invited (NPP Long-term Operation) 08:30-09:00 | Invited (Future Nuclear Plants) 08:30-09:00 |
| | Nuclear New Builds, Energy Policy and Decarbonization of Society 9:00-10:20 | Plant Life-time Extensions, Reliability, Outage Management, Innovations and Modernization 9:00-10:20 | Nuclear Fusion and Plasma Technologies 9:00-10:20 |
| The exhibition space layout (Sponsor exhibitors) 10:00-13:00 | Poster session @ Coffee break | Poster session @ Coffee break | Poster session @ Coffee break |
| | Nuclear Regulation, Society and Environment 11:00-12:30 | PSA and Severe Accidents 11:00-12:30 | Thermal Hydraulics and CFD 11:00-12:30 |
| Registration 13:00-14:45 | Lunch | Lunch | Lunch |
| | Nuclear Materials 13:45-15:00 | Nuclear Education, Training, Workforce Planning, Leadership and Talent Development 13:45-15:00 | |
| Opening ceremony 15:00-15:30 | Coffee break | Coffee break | |
| Invited (Role of Nuclear Energy in Decarbonization and National Energy Strategies) 15:30-17:00 | Thermal Hydraulics and CFD 15:20-16:20 | Reactor Physics and Research Reactors 15:20-16:20 | |
| Coffee break | Break | Break | |
| Round table (Capacity Building for Nuclear Energy Deployment and Utilization) 17:20-19:10 | Fuel Cycle and Radioactive Waste 16:30-18:00 | Reactor Physics and Research Reactors 16:30-18:15 | |
| Break | | | |
| Welcome reception 19:30-21:00 | Social event | Conference dinner 19:30-21:00 | |

## Program Highlights

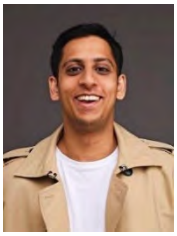
*Monday, September 11, 2023*

**Role of Nuclear Energy in Decarbonization and National Energy Strategy**

Yves Desbazeille

Director General

Nucleareurope, Belgium



**The UK's Approach Nuclear To Building The Evidence Base For Nuclear**

Mohammed Zaid Khonat

Head of Advanced Nuclear Policy and Delivery

Department for Energy Security and Net Zero, United Kingdom

**Energy policy**

Jean-Jacques Coursol

Head of French Long-Term Economics and Prospective

EDF Strategy Division, France

*Round table*

Andrew Worrall

Section Head of the Integrated Fuel Cycle Section

Oak Ridge National Laboratory

Leon Cizelj
President of the European Nuclear Society (ENS)

Karen Daifuku
Executive Director of the I2EN

Aleshia Duncan
Deputy Assistant Secretary, U.S. Department of Energy

Shin Whan Kim
Head of the Nuclear Power Engineering Section, IAEA

Beccy Pleasant
Head of Nuclear Skills Strategy, Nuclear Skills Strategy Group

Igor Sirc
Director at Slovenian Nuclear Safety Administration (SNSA)

## *Tuesday, September 12, 2023*

**Status and Development of JEK2 Project**

Bruno Glaser

Head of Technical Division

GEN energija, Slovenia

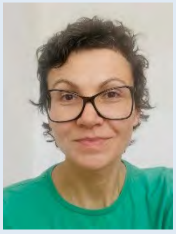
## *Wednesday, September 13, 2023*

**Environmental Impact Assessment Procedure for Krško NPP Lifetime Extension – Experience and Lessons Learned**

Aleksandra Antolovič

Analysis and Licensing Superintendent

Nuklearna elektrarna Krško, Slovenia



Ilijana Iveković

Project Leader in Technical Support Organisation

ENCONET, Croatia

## *Thursday, September 14, 2023*

**An outlook on future micro modular nuclear reactors**

Joerg Starflinger

Director of the Institute of Nuclear Technology and Energy Systems (IKE)

University of Stuttgart, Germany

# Conference Program

## *Monday, September 11*

| | | | |
|---|---|---|---|
| | | **15:00** | **Opening Ceremony** |

***Tomaž Žagar** – President of Nuclear Society of Slovenia*
***Leon Cizelj** – President of European Nuclear Society*
***Gorazd Pfeifer** – President of the Management Board of NEK*
***Dejan Paravan** – Chief Executive Officer of GEN energija*
***Representative of the Government of the Republic of Slovenia***

| | | | |
|---|---|---|---|
| Invited lectures | | **15:30** | **Role of Nuclear Energy in Decarbonization and National Energy Strategies** |
| 15:30 | *1* | ***Yves Desbazeille** – France*<br>**Role of Nuclear Energy in Decarbonization and National Energy Strategy** | |
| 16:00 | *2* | ***Mohammed Zaid Khonat** – United Kingdom*<br>**The UK's Approach Nuclear To Building The Evidence Base For Nuclear** | |
| 16:30 | *3* | ***Jean-Jacques Coursol** – France*<br>**Decarbonisation of the French power system in 2050 – complementarity between nuclear power and renewable energies** | |

| | | | |
|---|---|---|---|
| Round table | | **17:20** | **Capacity Building for Nuclear Energy Deployment and Utilization** |

*Speakers:*
*Andrew Worrall – United States of America*
*Leon Cizelj – Slovenia*
*Karen Daifuku – France*
*Aleshia Duncan – United States of America*
*Shin Whan Kim - IAEA*
*Beccy Pleasant – United Kingdom*
*Igor Sirc – Slovenia*

## *Tuesday, September 12*

| Invited lecture | 08:30 | **Status and Development of JEK2 Project** |
|---|---|---|

*Chairpersons:* *Martin Novšak*
*Janez Gale*

08:30 *4* ***Bruno Glaser** – Slovenia*
**Status and Development of JEK2 Project**

| Session 1 | 09:00 | **Nuclear New Builds, Energy Policy and Decarbonization of Society** |
|---|---|---|

*Chairpersons:* *Martin Novšak*
*Janez Gale*

09:00 *101* ***Roman Romanowski** – United States of America*
**Westinghouse in Focus: Navigating Market Dynamics with a Diverse Technology Portfolio**

09:15 *102* ***Anne Falchi** – France*
**EDF's approach to deliver Europe's and Slovania's nuclear ambition**

09:30 *103* ***Keunho Lee** – Republic of South Korea*
**Supply Chain Development and Localization Strategies for the New Nuclear Power Plant Krško in Slovenia**

09:45 *104* ***Gerard Cognet,** Michel Debes, Jan Bartak – France*
**New generations of nuclear reactors for flexible energy capacities: from load following to multi-energy capabilities**

10:00 *105* ***Rosa Lo Frano,** R. Buzzetti, Salvatore Angelo Cancemi – Italy*
**Preliminary feasibility study of coupling between SMR and desalination plant**

| Session 2 | 11:00 | **Nuclear Regulation, Society and Environment** |
|---|---|---|

*Chairpersons:* *Franck Wastin*
*Andreja Peršič*

11:00 *201* ***John Kickhofel**, Zoran Heruc – Switzerland*
**Use of Commercial grade items in nuclear safety related applications: New developments and guidance**

11:15 *202* ***Zdravka Škugor Ferdebar**, Krešimir Trontl, Mario Matijević – Croatia*
**Correlation Between Public Acceptance of Nuclear Technology and Trust in Scientists – Case Study Croatia**

11:30 *203* ***Lingteng Kong,** Dave Megson-Smith – United Kingdom*
**Advancements in Remote Monitoring of Alpha Emitters Using Alpha Induced Radio-luminescence**

11:45 *204* ***Vesna Kolar Planinšič -** Slovenia*
**Application of Espoo Convention on environmental impact assessment in transboudary context on Extension of life-time for nuclear power plant Krško**

12:00 *205* ***Tinkara Bučar,** Matjaž Stepišnik, Matjaž Koželj – Slovenia*
**Evaluation of the Justification for Using Consumer Products and Geological Samples Containing Radioactive Substances**

| | | |
|---|---|---|
| 12:15 | *206* <br> *Y.A.* | ***Vincenzo Anthony Di Nora**, Andreas Wielenberg, Thorsten Hollands – Germany* <br> **Implementation and application of a GitLab CI-pipeline in support of V&V activities of AC2** |
| 12:20 | *207* <br> *Y.A.* | ***Sebastian Thomas Oakes** – United Kingdom* <br> **Evolutionary computing for the estimation of radiation fields** |

## Session 3 **13:45** **Nuclear Materials**

*Chairpersons:* *Rosa Lo Frano*
*Leon Cizelj*

| | | |
|---|---|---|
| 13:45 | *301* | ***Chris Hutson** – United Kingdom* <br> **GANEX gamma radiation tolerance methodology** |
| 14:00 | *302* | ***Timon Mede**, Samir El Shawish – Slovenia* <br> **Probabilistic assessment of induced intergranular stresses in polycrystalline materials** |
| 14:15 | *303* | ***Daniela Marušáková,** Monika Šípová, Michal Novák, Radek Novotný – Czech Republic* <br> **Stress corrosion cracking behavior of stainless steel 310S in supercritical water** |
| 14:30 | *311* | ***Bojan Zajec**, Tim Austin, Rik-Wouter Bosch, Mary Grace Burke, Michael Grimm, Matthias Herbst, Anna Hojna, Tadeja Kosec, Andraž Legat, Agostino Maurotto, Jill Meadows, Radek Novotny, Valentin Olaru, Thomas Pasutto, Francisco Javier Perosanz Lopez, Zaiqing Que, Stefan Ritter, Veronica Román Flórez, Alberto Sáez Maderuelo, Fabio Scenini, Aki Toivonen, Aleksandra Treichel, Marc Vankeerberghen, Liberato Volpe, Krystian Wika, Mariia Zimina – Slovenia* <br> **Effect of surface machining on the environmentally-assisted cracking of Alloy 182 and 316L stainless steel in light water reactor environments – results of the collaborative project MEACTOS** |
| 14:45 | *305* <br> *Y.A.* | ***Jarred Minards** – United Kingdom* <br> **Characterisation of Graphite using Raman Spectroscopy** |
| 14:50 | *307* <br> *Y.A.* | ***Olga Vilkhivskaya,** Artem Lunev – Russian Federation* <br> **Enabling accurate characterisation of thermal properties in nuclear materials with an open-source software suite PULsE** |
| 14:55 | *308* <br> *Y.A.* | ***Amirhossein Lame Jouybari,** Samir El Shawish, Leon Cizelj – Slovenia* <br> **Analysis of Strain Localization in Irradiated Austenitic Stainless Steels using FFT-based Crystal Plasticity Simulations** |

## Session 4 I **15:20** **Thermal Hydraulics and CFD**

*Chairpersons:* *Joerg Starflinger*
*Matej Tekavčič*

| | | |
|---|---|---|
| 15:20 | *401* | ***Andrej Prošek,** Boštjan Končar – Slovenia* <br> **TRACE Analysis of Total Loss of Feedwater in PWR** |
| 15:35 | *408* <br> *Y.A.* | ***Žiga Perne,** Blaž Mikuž – Slovenia* <br> **Experimental Investigation of Taylor Bubble Decay Rate in Counter-current flow** |
| 15:40 | *409* <br> *Y.A.* | ***Martina Di Gennaro** – Italy* <br> **Sub-channel analysis of the MYRRHA reactor with OpenFOAM: Numerical assessment of turbulence models** |

| | | |
|---|---|---|
| 15:45 | *412*<br>*Y.A.* | ***Alessandro De Angelis**, Clément Liegeard, Walter Ambrosini – Italy*<br>**Sensitivity analyses by a CFD model on water-wall behaviour in a LW SMR during SBO conditions** |
| 15:50 | *413*<br>*Y.A.* | ***Nejc Kromar**, Oriol Costa Garrido, Leon Cizelj – Slovenia*<br>**Development of a 3D Extended Finite Element Model for Efficient Fracture Mechanics Analyses of Hollow Cylinders with Cracks** |
| 15:55 | *414*<br>*Y.A.* | ***Jan Kren**, Iztok Tiselj, Blaž Mikuž – Slovenia*<br>**Bubble Breakup Sensitivity on Local Surface Tension Modification in LES of Turbulent Slug Flow** |
| 16:00 | *415*<br>*Y.A.* | ***Aljoša Gajšek**, Matej Tekavčič, Boštjan Končar – Slovenia*<br>**A mechanistic bubble force model for the development of boiling parameters in high heat flux regimes** |

## Session 5 **16:30** **Fuel Cycle and Radioactive Waste**

*Chairpersons:* *Sean Michael Tyson*
*Krešimir Trontl*

| | | |
|---|---|---|
| 16:30 | *501* | ***Vid Merljak**, Rok Bizjak, Andrej Kavčič – Slovenia*<br>**Krško NPP Spent Fuel Dry Storage Project – Technology and Implementation Overview** |
| 16:45 | *502* | ***Charles McCombie**, Neil Chapman, Leon Kegel, Jake Kinghorn-Mills, Sean Tyson – Switzerland*<br>**How will backend issues affect the global deployment of SMRs?** |
| 17:00 | *503* | ***Stanko Manojlović**, Matjaž Gričar – Slovenia*<br>**Radwaste handover project to ARAO, Fond and Decommissioning of the original steam generators** |
| 17:15 | *504* | **Sebastian Schoop**, *Mehmet Kadiroglu, Kai-Martin Haendel – Germany*<br>**Calculation method for determining neutron-induced nuclide activities in nuclear facilities** |
| 17:30 | *505* | ***Špela Mechora**, Sandi Viršek – Slovenia*<br>**Updates on LILW disposal facility in Slovenia** |
| 17:45 | *507*<br>*Y.A.* | ***Philip Hutchinson**, Ross Springell, Tom Scott – United Kingdom*<br>**Characterisation of Uranium Metal Encapsulated in Magnox Sludge** |
| 17:50 | *508* | ***Thomas Heinz Wiese**, Jürgen Michna, Yannick Lotaut, Christophe Jarousse, Tony Pugh – Germany*<br>**Recent Developments Regarding Fuel Services Technologies With Focus On Spent Fuel Management** |

## *Wednesday, September 13*

| | | |
|---|---|---|
| Invited lecture | **08:30** | **Environmental Impact Assessment Procedure for Krško NPP Lifetime Extension – Experience and Lessons Learned** |

*Chairpersons:* *Didier Banner*
*Bruno Glaser*

| | | |
|---|---|---|
| 08:30 | *5* | ***Aleksandra Antolovič, Ilijana Ivеković*** *– Slovenia*<br>**Environmental Impact Assessment Procedure for Krško NPP Lifetime Extension – Experience and Lessons Learned** |

| | | |
|---|---|---|
| Session 6 | **09:00** | **Plant Life-Time Extensions, Reliability, Outage Management, Innovations and Modernization** |

*Chairpersons:* *Didier Banner*
*Bruno Glaser*

| | | |
|---|---|---|
| 09:00 | *601* | ***Joseph Redmond*** *– United States of America*<br>**Risk Informed Engineering Applications to reduce Operating and Maintenance Costs** |
| 09:15 | *602* | ***Marie Berthelot*** *– France*<br>**The MAI: An unique international collaborative research center dedicated to materials ageing for nuclear power plants** |
| 09:30 | *603* | ***Robert Planinc*** *– Slovenia*<br>**Mechanical Stress Improvement Process on NPP Krško Reactor Vessel Nozzles** |
| 09:45 | *604* | ***Daniel Celarec,*** *Milan Moškon, Alessandro Blascovich – Slovenia*<br>**On the reasons and extent of NEK's seismic instrumentation upgrade** |
| 10:00 | *605* | ***Shanqi Song****, David Megson-Smith, Matthew Ryan Tucker, Yannick Verbelen, Tom Scott – United Kingdom*<br>**Enhanced Indoor Inspection of Nuclear Facilities through Non-Rigid Airship-Based SLAM System** |

| | | |
|---|---|---|
| Session 7 | **11:00** | **PSA and Severe Accidents** |

*Chairpersons:* *Mitja Uršič*
*Eileen Langegger*

| | | |
|---|---|---|
| 11:00 | *701* | ***Cristina Dominguez****, Tatiana Taurines, Georges Repetto, Emmanuel Rouge – France*<br>**Fuel Cladding Deformation during LOCA: Comparison between Single-rod and Multi-rod tests** |
| 11:15 | *702* | ***Nejc Kromar****, Oriol Costa Garrido, Andrej Prošek, Leon Cizelj – Slovenia*<br>**Development of 3D fracture mechanics submodels for PTS analyses of an RPV with cracks** |
| 11:30 | *703* | ***Mike Kuznetsov****, Alexei Kotchourko, Wolfgang Breitung – Germany*<br>**The Scaling of Turbulent Flame Acceleration and Detonation Transition for Hydrogen-Air mixtures in the RUT Facility** |
| 11:45 | *704* | ***Liviusz Lovasz****, Peter Pandazis – Germany*<br>**Simulation of an SBO accident in a PWR with AC2/ATHLET-CD with and without using ATF cladding material** |
| 12:00 | *705* | ***Matjaž Dolšek*** *– Slovenia*<br>**Seismic risk assessment of safety-related SSCs** |

| | | |
|---|---|---|
| 12:15 | *706* <br> *Y.A.* | ***Seyed Ali Hosseini**, Francesco D'Auria – Italy* <br> **Probabilistic Risk Assessment is Conservative or Best-Estimate Approach?** |
| 12:20 | *707* <br> *Y.A.* | ***Caroline Denier**, Emmanuel de Bilbao, Jules Delacroix, Pascal Piluso – France* <br> **Experimental Measurements of Thermophysical Properties of Several Corium Compositions and Influence on Fuel-Coolant Interaction** |

## Session 8 **13:45** Nuclear Education, Training, Workforce Planning, Leadership and Talent Development

*Chairpersons:* *Romana Jordan*
*Gabriel Lazaro Pavel*

| | | |
|---|---|---|
| 13:45 | *801* | ***Gabriel Lazaro Pavel**, Karen Daifuku, Michèle Coeck, Csilla Pesznyak, Christian Schonfelder, Leon Cizelj – Belgium* <br> **Impact of the ENEN2plus project after one year of implementation** |
| 14:00 | *802* | ***Matjaž Žvar** – Slovenia* <br> **Capture, transfer and dissemination of knowledge at NEK** |
| 14:15 | *803* | ***Barbara Somfai**, Zoltán Hózer, Margit Fábián – Hungary* <br> **Nuclear research activities from the front to the back end at Centre for Energy Research, Budapest, Hungary** |
| 14:30 | *804* | ***Joerg Starflinger**, Markus Hofer, Roberta Cirillo, Gabriel Pavel – Germany* <br> **Towards Optimized Use of Research Reactors in Europe - Summary and Results** |
| 14:45 | *805* | ***Yannick Le Gonidec** – France* <br> **How to attract new talents in nuclear industry** |

## Session 9 I **15:20** Reactor Physics and Research Reactors

*Chairpersons:* *Radek Škoda*
*Timothy Valentine*

| | | |
|---|---|---|
| 15:20 | *901* | ***Christophe Destouches**, Luka Snoj, Loic Barbot, Vladimir Radulović, Nicolas Thiollay – France* <br> **Update on the review of more than 15 years of JSI – CEA fruitful collaboration on nuclear instrumentation developments** |
| 15:35 | *902* | ***Yohannes Molla**, Lydie Giot, David Laks – France* <br> **Uncertainty Propagation of decay data for Decay Heat Calculations: A Monte Carlo Approach** |
| 15:50 | *903* | ***Federico Alfinito**, Luca Morselli, Antonietta Donzella, Alberto Arzenton, Mattia Asti, Silva Bortolussi, Stefano Corradetti, Giancarlo D'Agostino, Marco Di Luzio, Matteo Ferrari, Andrea Gandini, Marianna Tosato, Valerio Di Marco, Marcello Lunardon, Valerio Villa, Andrea Salvini, Lisa Zangrando, Aldo Zenoni, Alberto Andrighetto – Italy* <br> **The ISOLPHARM project: 111Ag Production and separation at L.E.N.A.** |
| 16:05 | *904* | ***Stefano Riva**, Lorenzo Loi, Carolina Introini, Antonio Cammi, Xiang Wang – Italy* <br> **FEniCSx-OpenMC Coupling for Neutronic Calculation with Temperature Feedback** |

| Session 9 II | 16:30 | **Reactor Physics and Research Reactors** |
|---|---|---|

*Chairpersons:* *Christophe Destouches*
*Luka Snoj*

| | | |
|---|---|---|
| 16:30 | *905* | ***Carolina Introini,*** *Marco Herbas Lopez, Carlo Lombardi, Antonio Cammi – Italy*<br>**An Innovative Reactivity Control Strategy for Small Modular Reactors** |
| 16:45 | *906* | ***Klemen Ambrožič****, Valentina Sola, Igor Mandić, Marco Ferrero, Gregor Kramberger, Luka Snoj – Slovenia*<br>**Fast neutron fluence profiling at the JSI TRIGA reactor irradiation facility** |
| 17:00 | *907* | ***Maria Angela de Barros Correia Menezes****, Rodrigo Reis Moura, Radojko Jaćimović – Brazil*<br>**Influence of the temporal variability of neutron fluxes in the carousel of TRIGA Mark IPR-R1 research reactor, CDTN, Brazil, in the k0-method of neutron activation analysis** |
| 17:15 | *908*<br>*Y.A.* | ***Danna Zhou*** *– China*<br>**Influencing factors of Oxygen control performance of pt/air electrochemical oxygen pump in liquid** |
| 17:20 | *909*<br>*Y.A.* | ***Alejandro Marro*** *– Slovenia*<br>**Possibility of using so-called "binary" "grey" control rods for power regulation of a nuclear reactor** |
| 17:25 | *911*<br>*Y.A.* | ***Julijan Peric****, Vladimir Radulović, Luka Snoj – Slovenia*<br>**Characterization of Cherenkov Radiation for Nuclear Power Measurements: A Study at the JSI TRIGA Research Reactor** |
| 17:30 | *913*<br>*Y.A.* | ***Anže Pungerčič****, Alireza Haghighat, Luka Snoj – Slovenia*<br>**Experimental and Computational Validation of Novel Depletion Algorithm in the RAPID Code System using JSI TRIGA reactor** |
| 17:35 | *914*<br>*Y.A.* | ***Dániel Sebestény****, István Panka, Bálint Batki – Hungary*<br>**Burnup-dependent group constant parametrization by applying different machine learning methodologies** |
| 17:40 | *915*<br>*Y.A.* | ***Stefano Riva****, Carolina Introini, Antonio Cammi – Italy*<br>**Multi-Physics Model Correction with Data-Driven Reduced Order Modelling** |
| 17:45 | *916*<br>*Y.A.* | ***Lorenzo Loi****, Carolina Introini, Antonio Cammi – Italy*<br>**OpenMC Model Validation of the TRIGA Mark II Reactor** |
| 17:50 | *917*<br>*Y.A.* | ***Diego Martin Cuellar Fernandez****, Gašper Žerovnik, Jan Malec – Slovenia*<br>**Feasibility study for design and utilization of a cold neutron irradiation facility at the JSI TRIGA reactor** |
| 17:55 | *918*<br>*Y.A.* | ***Muhammad Zaki Abbas Awan****, Gašper Žerovnik, Marjan Kromar – Italy*<br>**An Analytical Model of Heat Transfer in a Fuel Rod Suitable for Neutron Calculations** |

## *Thursday, September 14*

| Invited lecture | 08:30 | **An Outlook on Future Micro Modular Nuclear Reactors** |
|---|---|---|

*Chairpersons:* *Boštjan Končar*
*Saša Novak*

08:30 *6* ***Joerg Starflinger*** *– Germany*
**An Outlook on Future Micro Modular Nuclear Reactors**

| Session 10 | 09:00 | **Nuclear Fusion and Plasma Technologies** |
|---|---|---|

*Chairpersons:* *Boštjan Končar*
*Saša Novak*

09:00 *1001* ***Sabina Markelj****, Janez Zavašnik, Andreja Šestan, Thomas Schwarz-Selinger, Mitja Kelemen, Esther Punzon Quijorna, David Dellasega, Mateo Passoni – Slovenia*
**The influence of grain size on the displacement damage creation, D retention and transport in tungsten**

09:15 *1002* ***Andreas Ikonomopoulos****, Mitja Uršič, Leon Cizelj, Juan Carlos de la Rosa Blul, Spyros Andronopoulos, Nicholas Terranova, Gian-Luigi Fiorini, Sigitas Rimkevičius, Egidijus Urbonavičius, Ismo Karppinen – Greece*
**Licensing fusion facilities based on the ITER and DEMO paradigms**

09:30 *1003* ***Vladimir Krsjak,*** *Pavol Noga, Stanislav Sojak, Yamin Song, Jarmila Degmova – Slovak Republic*
**Experimental simulation of fusion-relevant radiation environments in bulk samples**

09:45 *1004* ***Olga Ogorodnikova****, Mitja Majerle, Jakub Cizek, Peter Hruska, Stanislav Simakov, Milan Stefanik, Václav Zach – Russian Federation*
**Simulation of fusion neutron damage in tungsten and iron using high energy protons, high energy ions, high energy neutrons and fission neutrons**

10:00 *1005 Y.A.* ***Patrik Tarfila****, Oriol Costa Garrido, Boštjan Končar – Slovenia*
**Development of a Thermo-mechanical Model for DTT PFU**

10:05 *1007 Y.A.* ***Norbert Wegrzynowski****, Tomas Martin, Thomas Bligh Scott – United Kingdom*
**Novel Methods of Isotopic Separation of Lithium 6**

| Session 4 II | 11:00 | **Thermal Hydraulics and CFD II** |
|---|---|---|

*Chairpersons:* *Iztok Tiselj*
*Ivica Bašič*

11:00 *402* ***Sinem Cevikalp Usta****, Michael Buck, Joerg Starflinger – Germany*
**Analysis of an innovative passive heat removal system for station blackout scenario**

11:15 *403* ***Nicolas Goreaud****, Emilie may, Adrien Collin de l'Hortet – France*
**Use of CFD for fuel handling and storage safety**

11:30 *404* ***Adam Kecek****, Ladislav Vyskocil – Czech Republic*
**Application of CFD and system codes to simulation of HERO-2 experiment**

11:45 *405* ***Boštjan Zajec****, Leon Cizelj, Boštjan Končar – Slovenia*
**Lagrangian simulation of flow boiling experiments in horizontal annulus**

12:00 *416* ***Raksmy Nop**, Alan Burlot, Guillaume Bois, Blaž Mikuž, Iztok Tiselj – France*
**The 3D DNS simulation of a Taylor bubble in counter-current flow with a turbulent wake using the Front-Tracking method in TrioCFD**

12:15 *407* ***Qingling Cai**, Vincenzo Zingales, Francesco Auria, Dominique Bestion, Klaus Umminger – China*
**Sensitivity studies on Reverse Natural Circulation in the PKL facility by using RELAP5 mod3.2mz**

## Posters

Poster sessions will be held on Tuesday, Wednesday, and Thursday from 10:20 to 11:00.

### Nuclear New Builds, Energy Policy and Decarbonization of Society

106 ***Jure Jazbinšek*** *– Slovenia*
**Benefits of multi-unit NPP projects during construction phase with full life-cycle cost assessment**

107 ***Kaja Zupančič****, Tomaž Ploj, Tomaž Žagar – Slovenia*
**Nuclear power plants as a solution for reducing carbon footprint and achieving sustainable energy system**

108 ***Ondřej Burian****, Radek Škoda, Hussein Abdulkareem Saleh Abushamah – Czech Republic*
**A study of integration of Liquid Air Energy Storage (LAES) technology to nuclear district heating facility TEPLATOR**

109 ***Marija Zlata Božnar****, Primož Mlakar, Boštjan Grašič, Luka Štrubelj, Klemen Debelak, Gianni Tinarelli, Daniela Barbero, Elisa Specchia – Slovenia*
**Nuclear Power Plant Cooling Tower Steam Emission Environmental Impact**

110 ***Aleš Janžovnik****, Damjana Pirc, Bruno Glaser – Slovenia*
**Challenges Concerning the Siting of Nuclear Facilities in the Case of NPP2 in Krško**

111 ***Mojca Planinc****, Aleš Kelhar – Slovenia*
**Geological and Seismological site Investigations for Building New Nuclear Power Plant**

112 ***Tomi Živko****, Tomaž Nemec – Slovenia*
**Role of Small Modular Reactors in the Future of Nuclear Power**

113 ***Jurij Kurnik****, Robert Bergant – Slovenia*
**Nuclear Power Plant Design Assessment**

115 ***Hussein Abdulkareem Saleh Abushamah****, Tomáš Kořínek, Radek Škoda - Czech Republic*
**Integration of Heat-only Small Modular Reactor with Thermally Driven Systems**

116 ***Jan Lokar****, Aleš Kelhar, Robert Bergant – Slovenia*
**Potential Of Using SMR Reactors At Coal-Fired Power Plant Locations**

117 ***Pia Fackovič Volčanjk****, Andrej Senegačnik – Slovenia*
**Thermal Energy Storage Combined With a Molten Salt Reactor**

### Nuclear Regulation, Society and Environment

206 ***Vincenzo Anthony Di Nora****, Andreas Wielenberg, Thorsten Hollands – Germany*
**Implementation and application of a GitLab CI-pipeline in support of V&V activities of AC2**

207 ***Sebastian Thomas Oakes*** *– United Kingdom*
**Evolutionary computing for the estimation of radiation fields**

208 ***Ivo Žarković****, Rudi Janežič– Slovenia*
**Changes in the field of Occupational Safety and Health (OSH) at Krško Nuclear Power Plant**

209 ***Andreas Ikonomopoulos****, Spyros Andronopoulos, Mirela Nitoi, Walter Klein-Heßling, Danilo Ferretto, Alain Flores y Flores, Mitja Uršič, Rok Krpan, Boštjan Zajec, Leon Cizelj, Timo Löher, Gian-Luigi Fiorini, Egidijus Urbonavičius, Kateryna Fuzik, Mykola Sapon, Ismo Karppinen, Francesco Lodi – Greece*
**On the applicability of the IAEA documentation to innovative reactors**

210 ***Janez Češarek*** *– Slovenia*
**Transport of Radioactive Material: "The Road to Success – is Always under Construction"**

211 ***Francesco Lodi****, Giacomo Grasso, Mirela Nitoi, Marin Constantin, Minodora Apostol, Marco Caramello, Alain Flores y Flores, Antonio Dambrosio, Guido Mazzini, Oleksandr Kukhotskyi, Anatolii Shyshuta, Mitja Uršič, Andrej Prošek, Boštjan Zajec, Spyros Andronopoulos, Andreas Ikonomopoulos – Italy*
**Design innovations and novel safety claims impacting power plant licensing**

## Nuclear Materials

305 ***Jarred Minards*** *– United Kingdom*
**Characterisation of Graphite using Raman Spectroscopy**

306 ***Ewan James Woodbridge****, Dean Connor, Yannick Verbelen, Peter Martin, Thomas Bligh Scott – United Kingdom*
**Prospecting of lithium deposits through aerial mapping of naturally occurring radioactive material (NORM) using CsI detectors**

307 ***Olga Vilkhivskaya****, Artem Lunev – Russian Federation*
**Enabling accurate characterisation of thermal properties in nuclear materials with an open-source software suite PULsE**

308 ***Amirhossein Lame Jouybari****, Samir El Shawish, Leon Cizelj - Slovenia*
**Analysis of Strain Localization in Irradiated Austenitic Stainless Steels using FFT-based Crystal Plasticity Simulations**

309 ***Severine Annaval*** *– France*
**Digital radiography: alternative RT technique for vessel inspection**

310 ***Elchin Huseynov****, Raisa Hakhiyeva – Azerbaijan*
**Differential thermal analysis of gamma-irradiated nano titanium carbide particles**

312 ***Rolando Calabrese*** *– Italy*
**Thermal creep of MOX fuel: a review of correlations**

313 ***Jana Rejkova****, Daniela Marusakova, Marie Kudrnova – Czech Republic*
**Corrosion behaviour of materials in SCW environment**

## Thermal Hydraulics and CFD

408 ***Žiga Perne****, Blaž Mikuž – Slovenia*
**Experimental Investigation of Taylor Bubble Decay Rate in Counter-current flow**

409 ***Martina Di Gennaro*** *– Italy*
**Sub-channel analysis of the MYRRHA reactor with OpenFOAM: Numerical assessment of turbulence models**

410 ***Walter Ambrosini****, Sara Kassem, Andrea Pucciarelli – Italy*
**Extrapolating Phenomena Observed with Carbon Dioxide to Supercritical Water Reactor Conditions**

411 ***Markus Petroff****, Michael Buck, Joerg Starflinger – Germany*
**Development of surrogate models for quick estimation on debris bed coolability**

412 ***Alessandro De Angelis****, Clément Liegeard, Walter Ambrosini – Italy*
**Sensitivity analyses by a CFD model on water-wall behaviour in a LW SMR during SBO conditions**

413 ***Nejc Kromar**, Oriol Costa Garrido, Leon Cizelj – Slovenia*
**Development of a 3D Extended Finite Element Model for Efficient Fracture Mechanics Analyses of Hollow Cylinders with Cracks**

414 ***Jan Kren**, Iztok Tiselj, Blaž Mikuž – Slovenia*
**Bubble Breakup Sensitivity on Local Surface Tension Modification in LES of Turbulent Slug Flow**

415 ***Aljoša Gajšek**, Matej Tekavčič, Boštjan Končar – Slovenia*
**A mechanistic bubble force model for the development of boiling parameters in high heat flux regimes**

416 ***Raksmy Nop**, Alan Burlot, Guillaume Bois, Blaž Mikuž, Iztok Tiselj – France*
**The 3D DNS simulation of a Taylor bubble in counter-current flow with a turbulent wake using the Front-Tracking method in TrioCFD**

417 ***Andrej Prošek** – Slovenia*
**Uncertainty and Sensitivity Analysis of Hot Leg Loss of Coolant Accident Simulated by RELAP5**

418 ***Adam Kecek**, Zbynek Parduba – Czech Republic*
**Simulation of PASI experiment on passive containment heat removal system with MELCOR and ATHLET**

419 ***Ivo Kljenak**, Md Arman Arefin – Slovenia*
**Direct Contact Condensation Induced Water Hammer Simulation Using Computational Fluid Dynamics Code**

420 ***Rok Krpan**, Ivo Kljenak, Andrej Prošek – Slovenia*
**RELAP5 Validation Against Experiment Performed in SIRIO Experimental Facility**

421 ***Tomas Korinek**, Jan Skarohlid – Czech Republic*
**Experimental Investigation of Thermal Wave Flow Meter for Molten Salt Applications**

422 ***Vesna Benčik**, Davor Grgić, Siniša Šadek – Croatia*
**Analysis of Steam Generator Tube Rupture (SGTR) Accident using RELAP5/MOD 3.3 and TRACE 5.0p5 Codes**

423 ***Matej Tekavčič**, Richard Meller, Benjamin Krull, Fabian Schlegel – Slovenia*
**Resolution-Adaptive Modelling in Nuclear Safety: Free Surfaces and Bubbles**

424 ***Andrej Prošek** – Slovenia*
**Loss of Coolant Accident with Total Failure of High Pressure Injection System in Two-loop PWR**

425 ***Mihael Boštjan Končar**, Blaž Mikuž, Matej Tekavčič – Slovenia*
**Measurement of Radiative Transport Properties of Water and Refrigerant R245fa**

426 ***Antoaneta Stefanova**, Pavlin Groudev – Bulgaria*
**Comparison of the main parameters behaviour in VVER 1000 during a MSLB accident for different fuel campaigns**

427 ***Martin Draksler**, Christian Bachmann, Boštjan Končar – Slovenia*
**EU DEMO upper port transfer cask CFD analysis**

428 ***Davor Grgić**, Paulina Dučkić, Siniša Šadek, Petra Strmečki – Croatia*
**Calculation of HI-STORM FW cooling by natural convection of the air**

429 ***Davor Grgić**, Paulina Dučkić, Radomir Ječmenica, Bojan Petrović – Croatia*
**Usage of Monte Carlo Code Serpent2 for Calculation of FHR Fuel Assembly**

## Fuel Cycle and Radioactive Waste

506 ***Billy Murphy*** *– United Kingdom*
**Development of a wall crawling robot for remote inspection**

507 ***Philip Hutchinson****, Ross Springell, Tom Scott – United Kingdom*
**Characterisation of Uranium Metal Encapsulated in Magnox Sludge**

509 ***Sebastian Schoop****, Kai-Martin Haendel, Alexander Mühle – Germany*
**Uncertainty of Fuel Assembly Burnup in Siemens/KWU PWRs**

510 ***Nikolaja Podgoršek Selič*** *– Slovenia*
**The challenge of NORM waste from the production of titanium dioxide in Cinkarna Celje, d.d.**

511 ***Janez Vodopivec****, Simona Sučić, Marko Kostanjevec – Slovenia*
**Challenges and Objectives in Managing Institutional Radioactive Waste in Slovenia**

512 ***Margit Fabian****, Istvan Tolnai, Otto Czompoly – Hungary*
**Characteristics of a Steel/Concrete Model System Under Repository Conditions**

513 ***Jungjoon Lee*** *– Republic of South Korea*
**Development of Verification Tool for the Review of Site Survey Result Obtained from MARSSIM Sign and WRS test**

514 ***Leon Kegel****, Sandi Viršek – Slovenia*
**Deep Borehole Disposal as an Alternative Disposal Option for Spent Fuel**

515 ***Anna Sears****, Vojtech Galek, Thi Nhan Nguyen, Quoc Tri Phung – Czech Republic*
**Direct conditioning of liquid organic radioactive waste into a geopolymer matrix**

516 ***Szabina Török****, Zsófia Kókai – Hungary*
**Caks and lifting equipment for remote handling of large highly radioactive parts of a spallation source**

517 ***Vojtěch Galek****, Anna Sears, Petr Pražák, Martin Vacek, Jan Hadrava, Andrea Santi, Eros Mossini – Czech Republic*
**Improved Geopolymers For Encapsulation Of Molten Salts From Thermal Treatment Processes**

## Plant Life-Time Extensions, Reliability, Outage Management, Innovations and Modernization

605 ***Shanqi Song****, David Megson-Smith, Matthew Ryan Tucker, Yannick Verbelen, Tom Scott – United Kingdom*
**Enhanced Indoor Inspection of Nuclear Facilities through Non-Rigid Airship-Based SLAM System**

606 ***Domen Zorko****, Matej Pleterski, Aleš Jevnik – Slovenia*
**RPV threaded hole lubrication device**

607 ***Severine Annaval*** *– France*
**LT Cam : a contactless surface inspection tool with high sensitivity**

608 ***Severine Annaval*** *– France*
**Beznau RPV Shell UT Inspection and Indications Follow Up**

609 ***Justin Božič****, Krešimir Gudek – Slovenia*
**Completion of the Fourth 10-year ISI Program with Emphasis on Reactor Vessel Condition**

610 ***Rudolf Prosen****, Špalj Srđan, Barbara Colo Vrdoljak – Slovenia*
**Implementation of the NEK PSR3 Project and Action Plan**

611 ***Nina Gartner***, *Miha Hren, Andraž Legat, Sokratis Iliopoulos, Wouter Van Eesbeeck, Lucas Si Larbi, Eric Lucet – Slovenia*
**Robotic non-destructive inspection of corrosion inside steel cylinder concrete pipes used in nuclear power plants**

612 ***Davor Grgić***, *Paulina Dučkić, Mario Matijević, Siniša Šadek – Croatia*
**The Assessment of Dose Rates during MPC Loading and Drying in frame of the Nuclear Power Plant Krško First SFDS Loading Campaign**

613 ***Peter Hruščák***, *Guillaume Hémery – Slovak Republic*
**Early Launch of Validation via an Evolving Engineering Simulator (ELVEES)**

614 ***Željko Rapljenović***, *Luka Posilović, Marko Budimir – Croatia*
**Modernizing Manual Ultrasound NDE Inspection of Pipes in Nuclear Power Plants with Machine Learning**

## PSA and Severe Accidents

706 ***Seyed Ali Hosseini***, *Francesco D'Auria – Italy*
**Probabilistic Risk Assessment is Conservative or Best-Estimate Approach?**

707 ***Caroline Denier***, *Emmanuel de Bilbao, Jules Delacroix, Pascal Piluso – France*
**Experimental Measurements of Thermophysical Properties of Several Corium Compositions and Influence on Fuel-Coolant Interaction**

708 ***Janez Kokalj***, *Mitja Uršič, Matjaž Leskovar – Slovenia*
**Modelling approach for premixing phase in combination of melt jet breakup and premixed layer formation of melt spread**

709 ***Anna-Elina Pasi*** *– Finland*
**Organic chemistry of tellurium in severe accident scenarios**

710 ***Thomas Breznik***, *Benjamin Zorko, Miha Mihovilovič – Slovenia*
**Analysis of analogue data transmission of dose rate measurements over the HAM amateur radio network in the event of an emergency and possible failure of other communication channels**

711 ***Robi Jalovec***, *Mirko Bevc, Arunas Bieliauskas, Amparo Giner, Kevin Honath, Laura Genutis – Slovenia*
**PWROG SAMG Validation at NEK**

712 ***Alexander Vasiliev*** *– Russian Federation*
**Advanced Analytical and Numerical Modelling of ATF FeCrAl and Cr-Coated Zr-Based Cladding High Temperature Oxidation in Steam Atmosphere**

713 ***Matjaž Leskovar***, *Mitja Uršič, Janez Kokalj – Slovenia*
**Uncertainty Analysis of Severe Accident Scenario in Krško NPP**

714 ***Yves Pontillona*** *– France*
**Fission Products Behaviour with Respect to the Source Term in Severe Aaccident Conditions**

## Nuclear Education, Training, Workforce Planning, Leadership and Talent Development

806 ***Tomaž Skobe*** *– Slovenia*
**Basics of Nuclear Technology Courses in Nuclear Training Centre Ljubljana**

807 ***Barcza István*** *– Hungary*
**When Is The Best Time for Configuration Management in Nuclear Industry?**

808 ***Nadja Železnik*** *– Slovenia*
**Knowledge Management activities in the EURAD programme**

809 ***Radko Istenič****, Igor Jenčič – Slovenia*
**Youngsters about Nuclear Energy – Year 2023 Poll**

810 ***Radko Istenič****, Igor Jenčič – Slovenia*
**Public Opinion about Nuclear Energy – A Comparison between Youngsters and General Population, Year 2023 Poll**

811 ***Vesna Slapar Borišek****, Igor Jenčič, Tomaž Skobe – Slovenia*
**Renovation of the permanent exhibition on nuclear technology**

812 ***Kateryna Piliuhina****, Angel Papukchiev, Kevin Zwijsen, Philippe Planquart – Belgium*
**Vibration impact in nuclear power generation: Go-viking advancements**

## Reactor Physics and Research Reactors

908 ***Danna Zhou*** *– China*
**Influencing Factors of Oxygen Control Performance of Pt/air Electrochemical Oxygen Pump in Liquid LBE**

909 ***Alejandro Marro*** *– Slovenia*
**Possibility of using so-called “binary” “grey” control rods for power regulation of a nuclear reactor**

910 ***Tanja Goričanec****, Luka Snoj, Marjan Kromar – Slovenia*
**Intermediate range detectors for control rod worth measurements with rod insertion method**

911 ***Julijan Peric****, Vladimir Radulović, Luka Snoj – Slovenia*
**Characterization of Cherenkov Radiation for Nuclear Power Measurements: A Study at the JSI TRIGA Research Reactor**

912 ***Lorenzo Loi****, Stefano Riva, Carolina Introini, Antonio Cammi, Enrico Padovani – Italy*
**OpenMC Analysis of TRIGA Mark II Reactor Void and Temperature Reactivity Coefficients**

913 ***Anže Pungerčič****, Alireza Haghighat, Luka Snoj – Slovenia*
**Experimental and Computational Validation of Novel Depletion Algorithm in the RAPID Code System using JSI TRIGA reactor**

914 ***Dániel Sebestény****, István Panka, Bálint Batki – Hungary*
**Burnup-dependent group constant parametrization by applying different machine learning methodologies**

915 ***Stefano Riva****, Carolina Introini, Antonio Cammi – Italy*
**Multi-Physics Model Correction with Data-Driven Reduced Order Modelling**

916 ***Lorenzo Loi****, Carolina Introini, Antonio Cammi – Italy*
**OpenMC Model Validation of the TRIGA Mark II Reactor**

917 ***Diego Martin Cuellar Fernandez****, Gašper Žerovnik, Jan Malec – Slovenia*
**Feasibility study for design and utilization of a cold neutron irradiation facility at the JSI TRIGA reactor**

918 ***Muhammad Zaki Abbas Awan****, Gašper Žerovnik, Marjan Kromar – Italy*
**An Analytical Model of Heat Transfer in a Fuel Rod Suitable for Neutron Calculations**

919 ***Sahar Shirzadi Deh Kohneh****, Andrius Slavickas., Tadas Kaliatka – Lithuania*
**Parametric Analysis of RBMK Fuel Depletion Calculation Using Scale Code**

920 ***Dušan Čalič*** *– Slovenia*
**Evaluation of the Krško NPP core using Monte Carlo approach**

921 ***Martin Lovecký****, Jiří Závorka, Jana Jiřičková, Radek Škoda – Czech Republic*
**Radiation Damage of SMR Reactor Vessel**

922 ***Dušan Čalič****, Gašper Žerovnik, Jan Malec, Samo Gerkšič, Andrej Trkov, Marjan Kromar, Tomaž Žagar, Anže Mihelčič, Blaž Levpušček, Luka Snoj – Slovenia*
**An overview of research project of the nuclear power plant load follow operation**

923 ***Anže Jazbec****, Sebastjan Rupnik, Vladimir Radulović, Borut Smodiš, Luka Snoj – Slovenia*
**Jožef Stefan Institute TRIGA Research Reactor Activities in the Period from September 2022 – August 2023**

924 ***Bojan Žefran****, Andrej Trkov, Slavko Slavič, Jan Malec, Sebastian Pleško – Slovenia*
**Computer Programs Developed at Reactor Physics Department**

925 ***Jiri Zavorka****, Martin Lovecky, Radek Škoda – Czech Republic*
**Calculation of Neutron and Electron Transport in Self-Powered Neutron Detector for PWR Fuel Assembly**

926 ***Dipanjan Ray****, Martin Lovecký, Jiří Závorka, Radek Škoda – Czech Republic*
**Neutronic modelling and computation of TEPLATOR core using COMSOL Multiphysics code**

927 ***Dipanjan Ray****, Martin Lovecký, Jiří Závorka, Tomáš Peltan, Eva Vilímová, Martin Šašek, Radek Škoda – Czech Republic*
**Development of multi-point kinetic model for TEPLATOR**

928 ***Ingrid Švajger****, Nina C. Fleming, Gilles Noguere, Ayman I. Hawari, Luka Snoj, Andrej Trkov – Slovenia*
**Thermal Scattering Law for Zirconium Hydride**

929 ***Jan Malec****, Andrej Trkov, Marjan Kromar – Slovenia*
**On the Dependence of Reactivity Loss With Burnup Due To Nuclear Data**

930 ***Rolando Calabrese****, Shun Hirooka – Italy*
**Evaluation of Uncertainties affecting Elastic Properties of MOX fuel**

931 ***Victor Kolykhanov*** *– Ukraine*
**Angular Spectrum of the Diffusion Neutron Flux**

932 ***Anna Talarowska*** *– Poland*
**The calorimeter design for low nuclear heating rates for the MARIA research reactor**

## Nuclear Fusion and Plasma Technologies

1005 ***Patrik Tarfila****, Oriol Costa Garrido, Boštjan Končar – Slovenia*
**Development of a Thermo-mechanical Model for DTT PFU**

1006 ***Diana Knyzhnykova****, Saša Novak, Aljaž Iveković – Slovenia*
**Optimization of Cu melt infiltration in a W skeleton for diverter material applications**

1007 ***Norbert Wegrzynowski****, Tomas Martin, Thomas Bligh Scott – United Kingdom*
**Novel Methods of Isotopic Separation of Lithium 6**

1008 ***Esther Punzón-Quijorna****, Miguel Luis Crespillo Almenara, Gastón García López, Xin Jin, Flyura Djurabekova, Kai Nordlund, Janez Zavašnik, Andreja Šestan, Thomas Schwarz-Selinger, Sabina Markelj – Slovenia*
**Multi-Energy Rutherford Backscattering Spectroscopy in Channeling configuration for the analysis of defects in tungsten**

1009 ***Leon Bogdanović***, *Sergey Pestchanyi, Leon Kos – Slovenia*
**Divertor Armor Damage Estimation and Dynamics of Secondary Tungsten Plasma Simulation with TOKES in ITER**

1010 ***Mitja Kelemen***, *Sabina Markelj, Aleksanda Cvetinović, Matej Lipoglavšek, Miha Čekada, Primož Pelicon, Mickael Payet, Christian Grisolia – Slovenia*
**Cross section data for the Tritium-Helium-3 nuclear reaction from 0.7 to 5.1 MeV**

1011 ***Alex Martin Little***, *Hugo Dominguez-Andrade, Norbert Wegrzynowski, Tom Wallace-Smith, Mahmoud Bakr Arby, Yannick Verbelen, Tomas Martin, Thomas Bligh Scott – United Kingdom*
**Production of high porosity lithium hydride pellets for a solid-state fusion breeder blanket concept**

1012 ***Jernej Kovačič***, *Tomaž Gyergyek, Stefan Costea, Leon Kos – Slovenia*
**Steady-state vs. bloby plasma source: implications on divertor**

1013 ***Yannick Verbelen***, *Mahmoud Bakr Arby, Tom Scott – United Kingdom*
**Radiation Tolerant Neutron Activation Detector for Compact Inertial Electrostatically Confined Fusors**

## *Table of Contents*

# *Invited lectures*

1

# Role of Nuclear Energy in Decarbonization and National Energy Strategy

Yves Desbazeille

Nucleareurope, France

yves.desbazeille@nucleareurope.eu

Role of Nuclear Energy in Decarbonization and National Energy Strategy: Europe is currently facing significant challenges (War in Ukraine, energy prices surge and inflation) that impact its climate and energy policy. While a few years ago its energy strategy was focusing on environment and climate concerns only, security of supply and affordability are now very high in agendas. This change of paradigm has a positive influence on nuclear energy with much more lenient and even supportive policies at national level across the EU. The European Commission is also taking a more positive approach on nuclear, even if it remains relatively timid for now, mainly due to some strong opposition from the antinuclear bloc.

A quick review of current policies and perspectives will also be presented.

**Speaker Bio**

Yves Desbazeille is French and graduated in electrical engineering from the Ecole Supérieure d'Electricité ("SUPELEC") in France in 1991 and studied on an MBA program in the early 2000s. During his successful career, he has been involved in different businesses and responsibilities at EDF: nuclear engineering, hydro and thermal power projects management in France, USA as well as in Asia, where he was for 5 years. His previous position as EDF representative for energy in Brussels has provided him with an in-depth knowledge of the EU institutions and Brussels' stakeholders and of the energy and climate stakes for Europe.

Nucleareurope is the Brussels-based trade association for the nuclear energy industry in Europe. FORATOM acts as the voice of the European nuclear industry in energy policy discussions with EU institutions and other key stakeholders. The membership of nucleareurope is made up of 15 national nuclear associations and 2 corporate members.

2

## The UK's Approach Nuclear To Building The Evidence Base For Nuclear

Mohammed Zaid Khonat

Department for Energy Security & Net Zero, London, United Kingdom

Mohammed.Khonat@beis.gov.uk

In 2022, the UK Government announced its ambition of generating up to 24GW of nuclear-sourced energy by 2050. Public support will be vital if the UK is to successfully achieve its ambition and tackle its current energy challenge of transforming its energy infrastructure into a system that maintains energy security, ensures energy prices are affordable, and reduces greenhouse gas emissions to help the global fight against the impacts of climate change. The UK nuclear sector has only recently experienced a change towards a more open and transparent approach to public engagement. Society's awareness, understanding and acceptance of developments in energy technologies including SMRs is vital in achieving the UK's goals of ensuring secure, affordable and low carbon energy for decades to come. The presentation will share the UK's approach to building the evidence base for Nuclear working with a diverse group of stakeholders.

**Speaker Bio**

Mohammed Zaid Khonat is an experienced, high performing senior policy professional with extensive domestic and diplomatic experience providing advice to senior stakeholders including the Prime Minister on governmental affairs. Currently, he is the Head of Advanced Nuclear Policy and Delivery at the Department for Energy Security and Net Zero, leading a team to deliver on the UK government's energy security and net zero agenda.

His team is developing policy for the delivery of small modular reactors (SMRs) and advanced modular reactors (AMRs) in the UK. His main responsibilities in the role include providing primary advice to senior stakeholders including the Prime Minister and Secretary of State on SMR and AMR policy. Additional responsibilities include being a key member of the Senior Leadership Team mentoring and developing staff across the Civil Service.

3

# Decarbonisation of the French power system in 2050 – complementarity between nuclear power and renewable energies

Jean-Jacques Coursol

EDF, France

jean-jacques.coursol@edf.fr

The French National Carbon-Free Strategy defined by public authorities sets the guidelines to reach carbon neutrality by 2050: almost halving the energy consumption by 2050, and increasing electricity demand, as electricity is a key driver for both decarbonisation and energy efficiency.

Thanks to its nuclear fleet and to renewables, in particular hydroelectricity, the French power mix is today 92% carbon-free. However important choices need to be made soon, especially on whether to build new nuclear power plants, for a sustainable supply of low-carbon electricity.

Entitled by the French authorities, the French TSO (RTE), after a 2-year consultation process involving all stakeholders, published the Futures Energétiques 2050, an in-depth study on the evolution of the power system. By studying a wide range of electricity mixes (including 100% renewables), RTE has demonstrated the many advantages of having a mix that sustainably combines renewables and nuclear power, not only from a technical, industrial and economic point of view, but also from a societal and environmental one.

## Speaker Bio

Graduated of École Centrale Paris, Jean-Jacques Coursol worked at RTE - French TSO - for a decade, both on operational experience in regional and national dispatching unit and in Electric System Economics Division before joining EDF R&D in 2015. Since 2019, he has been in charge of strategic and economic studies for France in EDF Strategy Division.

4

# Status and development of JEK2 Project

Bruno Glaser

GEN energija d.o.o., Slovenia

bruno.glaser@gen-energija.si

GEN energija, as an investor, is leading the JEK2 project, which is currently in the phase of preparation for siting procedure. The preparation of the material necessary for the start of the Site spatial plan, which is planned for the beginning of next year, is underway. Analyzes of the site investigation, which form the basis of the site safety analysis report, are underway. We analyze the financial and business models that enable the financing of such a project, where co-investment is also not excluded. Intensive technical dialog is ongoing with potential vendors. The key activities and focus of the work are aimed at the final investment decision.

**Speaker Bio**

Bruno Glaser was employed at NPP Krško (NEK) after completing his studies in electrical engineering. He completed his master's degree in electrical engineering, and his PhD studies in mechanical engineering. During his career he worked in the field of safety analysis and licensing and managed many projects at the NEK. For the last four years, he has been employed at the GEN company, where he works as the head of Technical Division and NPP Krško 2 (JEK2) project manager.

5

# Environmental Impact Assessment Procedure for Krško NPP Lifetime Extension – Experience and Lessons Learned

Aleksandra Antolovič[1], Ilijana Ivekovič[2]

[1]Nuklearna elektrarna Krško d.o.o., Slovenija

[2]Enconet d.o.o., Croatia

aleksandra.antolovic@nek.si

In accordance with well-established practice recognized globally, a decision was made to extend the lifetime of Krško NPP, from 40 to 60 years. In order to achieve this, Krško NPP had to meet a number of preconditions. One of the preconditions was the environmental impact assessment and environmental consent. The presentation describes the environmental impact assessment procedure that was carried out for Krško NPP lifetime extension and provides an overview of good practices and lessons learned in the process of public participation in Slovenia and in other European countries participating in the transboundary EIA procedure in accordance with the Espoo Convention.

## Speakers Bio

Aleksandra Antolovič

Aleksandra Antolovič graduated from the Faculty of Chemistry and Chemical Technology at the University of Ljubljana. She got a job at the Krško Nuclear Power Plant. She has overall 25 years of experience in various fields, such as nuclear fuel and reactor core, periodic safety review, safety analysis, and licensing. She managed some projects; one of the last being an environmental impact assessment for the lifetime extension of the Krško NPP.

Ilijana Ivekovič

Ilijana Iveković is a seasoned professional with 20 years of experience in conducting safety assessments for nuclear power plants and other nuclear facilities. Holds a Master's degree in Electrical Engineering and up-to-date knowledge of nuclear safety standards and best practices. After being employed at Faculty of Electrical Engineering, University of Zagreb, she joined ENCONET, a Technical Support Organisation, where she has contributed to various projects for Krško Nuclear Power Plant.

6

# An outlook on future micro modular nuclear reactors

Joerg Starflinger

University of Stuttgart, Germany

joerg.starflinger@ike.uni-stuttgart.de

Micro Modular Reactors (MMR) as a versatile and fast developing sub group within advanced small modular reactors (SMR) development.

Such MMR are innovative nuclear power plants for various applications, e.g. for both in space, like the Kilopower concept that NASA wants to put on moon or concepts that power satellites. For terrestrial use, such plants could be foreseen for remote areas with no existing connection to the electricity grid, mining or military applications. The use of very small reactors could also be beneficial in disaster areas by providing electricity (and optionally heat) for benefit of the affected people.

A general characteristic of these kind of reactors is the construction in factories and transport to the "site of use" instead of transport of all components to and construction on the "site of use". The technology readiness level is usually at "Conceptual Design" with some experimental for verification (TRL-3). Some reactor concepts use well-established technologies downscaled to small sizes.

The presentation contains an overview of these future innovative reactors and explains selected MMR concepts.

## Speaker Bio

Joerg Starflinger is the Director of the Institute of Nuclear Technology and Energy Systems (IKE) at University of Stuttgart. He is working on the assessment of innovative nuclear systems and development of technology to improve the safety of NPP. Current research activities include passive heat removal by means of heat pipes or thermosiphons and assessment of highly mobile micro modular reactors. Being a Professor and Head of the institute, he is teaching several lectures on nuclear technologies and nuclear safety.

Joerg has received his PhD in Mechanical Engineering from Ruhr-University Bochum. He spent 12 years at Nuclear Research Centre Karlsruhe (now KIT Campus North) as a Post-Doc, Research Scientist and Group leader. In 2012, he became Professor, Chair of Reactor Technology and Nuclear Safety, at University of Stuttgart. He is engaged in education and training on national and European level. He served as president of the European Nuclear Education Network (ENEN) and is Board Member of the German Nuclear Society.

## *Nuclear New Builds, Energy Policy and Decarbonization of Society*

101

## Westinghouse in Focus: Navigating Market Dynamics with a Diverse Technology Portfolio

Roman Romanowski

Westinghouse, United States of America

Roman.Romanowski@westinghouse.com

This presentation provides insights about Westinghouse's influential role in shaping the nuclear landscape and company's noteworthy accomplishments. It provides an understanding of the difficulties faced by the industry and highlights Westinghouse's strategic responses to these challenges as well as a summarized outlook of Westinghouse's future within evolving industry landscape. It serves to enrich the audience's understanding of Westinghouse's resilience and adaptability amid market changes, underpinned by its wide-ranging nuclear and non-nuclear technology portfolio.

---

102

## EDF's approach to deliver Europe's and Slovenia's nuclear ambition

Anne Falchi

EDF, France

anne.falchi@edf.fr

In the context of the energy crisis and the ambition to reach a net zero economy by 2050, many European countries 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. EDF together with the French and European nuclear industry, is providing a European solution to these challenges, based on four pilars: A european design, enabling each owner to retain its strategic independence; a European supply chain, with 95% of the value of our on-going projects in Europe; a 100% European solution for the fuel supply; A collaborative approach to deliver the best projects for Europe, by developing a portfolio of products designed for the European needs, from EPR to the SMR Nuward.

103

# Supply Chain Development and Localization Strategies for the New Nuclear Power Plant Krško in Slovenia

Keunho Lee

KHNP – Korea Hydro & Nuclear Power co. Ltd., Republic of South Korea

lkhs2000@khnp.co.kr

Korea, drawing upon its extensive experience in constructing and operating nuclear power plants over the past five decades, has cultivated a highly efficient supply chain ecosystem. The collaborative efforts among nuclear companies involved in design, construction, equipment manufacturing, fuel supply, project management, and operation have contributed to the establishment of a robust supply chain network. KHNP, a prominent entity in the Korean nuclear industry, has demonstrated exceptional project management, commissioning, and operational expertise through effective partnerships with KEPCO E&C, KNF, Doosan Enerbility, and competitive construction companies. Leveraging this wealth of experience and the availability of qualified supply chains, Korea possesses a distinct competitive advantage in undertaking new build projects globally.

The new NPP Krško project presents a significant opportunity to leverage local resources. By engaging local suppliers, labor, materials, and expertise, the project stands to benefit from improved efficiency and cost-effectiveness. In order to develop the supply chain, initial surveys are conducted by subcontracting the supply chain investigation and by directly sourcing through local networks. Companies that have successfully passed KHNP's qualification process are then registered on the AVL (Approved Vendor List). Throughout this process, KHNP intends to provide full support to local companies in order to maximize their registration.

The participation of local vendors is a prerequisite for the construction and operation of a nuclear power plant. To ensure the commercial success of the project, it is essential to effectively utilize local resources. Both the owner and the supplier can reduce overall project costs by harnessing these local resources. Therefore, KHNP is committed to maximizing localization efforts to promote the advancement of Slovenia's nuclear power industry.

104

# New generations of nuclear reactors for flexible energy capacities: from load following to multi-energy capabilities

Gerard Cognet, Michel Debes, Jan Bartak

NucAdvisor, France

gerard.cognet@nucadvisor.com

Nuclear reactors are an essential part of the global energy mix of a number of industrialized countries throughout the world. In Europe, even though several countries have chosen nuclear phase out, nuclear energy provides a significant proportion of electricity needs. However, as decarbonization of energy production and transport is a priority, the role of nuclear in energy strategies is changing.

As regards electricity production, variable renewable energy sources (RES) such as wind and solar power can contribute to decarbonization by displacing fossil fuels; therefore, their deployments increase. However, a large installed capacity of intermittent RES can cause sudden fluctuations in power supply and threaten grid stability. This is particularly an issue in systems with a high share of nuclear power that typically operated in base load. Moreover, since in this case one source of low-carbon electricity displaces another one, the contribution to emissions reduction is questionable, while the threat to grid stability is a real problem. Consequently, instead of operating predominantly in base load, nuclear generators have to contribute to the flexibility of the electrical system by adapting their production both to the variation of demand and to the variable production of intermittent renewables. This capability, known as load following, is critical to maintaining grid stability and preventing blackouts in systems with a high penetration of variable RES.</p>
<p>From the beginning of the civil use of nuclear energy, the production of heat was envisaged in parallel with the production of electricity, using steam extracted from the turbine. This type of cogeneration has even been widely implemented in some countries. Today, different modes of electricity-heat cogeneration are proposed, and certain reactors are designed for the production of heat for various industrial applications, for district heating or desalination.

Hydrogen is currently considered as a promising path for the decarbonization of the hard-to-abate industrial and transport sectors. In this perspective, nuclear energy could contribute to massive production of carbon-free hydrogen by using electricity, or heat, or both.

Flexibility remains a key requirement in hybrid energy systems with diversified sources having very different production characteristics. In this regard, SMRs (Small Modular Reactors) and certain advanced large power reactor designs offer greater flexibility in their power output compared to current generation of large reactors. They can be designed to operate at a range of power levels, making them better suited for load following and grid integration with the highly variable renewable energy sources. Some SMRs are developed for specific applications, such as HTRs for electricity-hydrogen cogeneration or high-temperature heat production for the steel industry. Another innovative solution announced for certain SMRs consists in incorporating energy storage which provides a new type of flexibility.

In Europe, we observe the emergence of many SMR projects whose flexibility is widely put forward to meet industrial needs currently covered by thermal power plants. This is even more true in France where a competition has been set up by the government which plans to subsidize the most promising projects. Many of these projects, both in France and in other European countries, are carried out by start-up companies, which considerably modifies both the industrial and financial approach.

In this paper, after a review of the constraints imposed by flexibility and in particular load following, we will discuss the proposed solutions by nuclear energy to meet the requirements of multi-energy production and what are the limitations related to nuclear safety.

105

## Preliminary feasibility study of coupling between SMR and desalination plant

Rosa Lo Frano, R. Buzzetti, Salvatore Angelo Cancemi

University of Pisa, Italy

rosa.lofrano@ing.unipi.it

The aim of the paper is to assess the feasibility of nuclear desalination, which will be obtained using both electricity and heat generated by nuclear power plant to remove salt and minerals from seawater. The integration of a water desalination plant into a small and modular nuclear power plant is described by considering a combination of a variety of seawater desalination co-generation configurations/ techniques (thermal or membrane in single or hybrid mode) to show they are successfully coupled with SMRs (of different types) to produce water and electricity at different scales.

Running SMRs as base load plants is more economical and simpler than requiring them to follow load. Therefore, in a cogeneration mode and while grid load is low, they may run at full capacity even if their capacity exceeds water demands.

The proposed solution was numerically investigated from both thermodynamic and economic points of view using the Desalination Economic Evaluation Program (DEEP) software made available by the IAEA.

The study highlights the role of factors such as site characteristics, plant capacity, feed/product-water quality, energy costs, in affecting the economics of desalination regardless of the energy source used.

The economics of nuclear desalination has been found to be competitive with other desalination techniques driven by other sources of energy. Results shown that e.g., for an average per capita electricity consumption of 4.7 MWh/year and 80.3 $m^3$/year of water, the CAREM25 reactor coupled to a desalination plant could produce electricity for 35,000 inhabitants and water for domestic use for 200,000 inhabitants.

106

## Benefits of multi-unit NPP projects during construction phase with full life-cycle cost assessment

Jure Jazbinšek

GEN energija d.o.o., Slovenia

jure.jazbinsek@gen-energija.si

Analysis of Overnight Construction Costs of various Nuclear Powerplant projects with single and multiple units on site.

Results expose benefits of multi-unit NPP projects, which statistically experiencing shorter construction times and overall lower costs of additional units on site.

Additional analysis of cost assessment for operational NPPs provide inputs for comprehensive view of full life-cycle assessments of NPPs with extended long term operation projections.

Result can be beneficial for future nuclear projects, where demanding financial decision for multi-unit NPPs can be additionally supported with past results, present technology advancements and future development projections.

107

## Nuclear power plants as a solution for reducing carbon footprint and achieving sustainable energy system

Kaja Zupančič, Tomaž Ploj, Tomaž Žagar

GEN energija d.o.o., Slovenia

kaja.zupancic@gen-energija.si

The paper will present the current requirements and practice of carbon footprint reporting, with the aim of reducing greenhouse gas emissions while following legislation and ESG (Environmental, Social, and Governance) framework. The practice of calculating the carbon footprint will be presented and, as an example, a general overview of the importance of nuclear facilities for reducing the carbon footprint will be given.

The calculation of carbon footprint is a new element of a comprehensive environmental impact management system, as it enables monitoring of progress and comparisons within similar activities. By calculating an accurate and verifiable carbon footprint, organizations identify their activities that are the most burdensome with greenhouse gas emissions. For many companies, purchased electricity presents a significant part of those emissions. and that is why organizations need to follow the trend of reducing emissions, resource

consumption and costs to achieve the greatest possible progress, both on a business and sustainable level area. In practice, customers or users, companies and banks demand more and more such statements from the companies they work with. One of the key indicators of an organization's efforts to transition to sustainable operations is also the timely preparation of a quality carbon footprint report.

All the above are reasons why companies want to reduce their carbon footprint. By switching to a fossil-free electricity therefore enables companies to lower its indirect emissions and show their dedication to be more environmentally friendly. The need for fossil-free electricity will therefore increase shortly and nuclear power plants are a solution to deliver such electricity.

108

## A study of integration of Liquid Air Energy Storage (LAES) technology to nuclear district heating facility TEPLATOR

Ondřej Burian, Radek Škoda, Hussein Abdulkareem Saleh Abushamah

University of West Bohemia, Faculty of Electrical Engineering, Czech Republic

buriano@fel.zcu.cz

The increase in the share of renewable energy sources in power production increases the need for energy storage technologies. These technologies are necessary for the reliable and stable operation of power grids for compensation of differences between energy production from renewables and energy consumption. At the same time, there is a lack of sustainable heat sources for district heating. In this case, a sustainable source is meant for zero-emission and low-cost heat production.

One of the ways for the integration of energy storage technologies into industrial-scale operation is in combination with other technologies for energy production. This article is focused on to study of the integration of two unique energy technologies into one unit. They are Liquid Air Energy Storage (LAES) technology, electricity storage, and technology for nuclear district heating generation TEPLATOR. This combination is very specific due to the combination of electricity storage large-scale technology and heat generation technology, but on the other hand, this combination promises an increase in energy production and storage compared with two single technologies. The article is introduced possible systems interconnections, operation operating modes, and critical technical-economic evaluation of the introduced system integration.

109

## Nuclear Power Plant Cooling Tower Steam Emission Environmental Impact

Marija Zlata Božnar[1], Primož Mlakar[1], Boštjan Grašič[1], Luka Štrubelj[2], Klemen Debelak[2], Gianni Tinarelli[3], Daniela Barbero[3], Elisa Specchia[3]

[1] MEIS d.o.o., Slovenia

[2] GEN energija d.o.o, Slovenia

[3] Arianet s.r.l., Italy

marija.zlata.boznar@meis.si

GEN energija is planning to build a new nuclear power plant in Slovenia. Due to limited heat sink capabilities of the Sava River cooling with wet cooling towers will have to be applied. A study was made in order to determine the environmental impact of the steam (plume) from cooling towers to the nearby and wider surroundings. The thermal power of analysed reactor was 3500 MW, electrical power of turbine was 1200 MW and the thermal power rejected into environment by cooling tower was 2300 MW. The analysed cooling tower was 175 m high with 72 m diameter. The steam plume dispersion was numerically simulated for every half hour in 2 years of real weather conditions. When air is oversaturated with moisture the steam is visible as cloud and can have an effect to environment. The timing, amount and duration of visible plume was analysed. The shadow from the plume and the plume height was calculated. Other effect such as increase in air humidity and effect to agriculture were recognized. The simulation showed that the environmental impact is only minor and is noticeable only in the vicinity of the cooling tower. Some effects are beneficial to environment such as increased mixing of the atmosphere and reduced fog at the ground; reduced frost at the ground; and reduced vaporization of water from the soil.

110

## Challenges Concerning the Siting of Nuclear Facilities in the Case of NPP2 in Krško

Aleš Janžovnik[1], Damjana Pirc[1], Bruno Glaser[2]

[1] SAVAPROJEKT d.d., Slovenia

[2] GEN energija d.o.o., Slovenia

savaprojekt@savaprojekt.si

The placement and construction of new nuclear facilities represent a great challenge. Even though 60 new nuclear power plants are currently being planned and built around the world, in Europe nuclear energy was not on the priority list until recently (WNA[1], 2023). The war in Ukraine and the related energy crisis have shown the vulnerability of energy system and revived considerations about the use of nuclear energy, so it is not surprising that the European Parliament classified nuclear energy as green energy in July 2022 (Reuters, 2022). This opened the door wide again for the increase of nuclear energy use. New nuclear power plants are being built in several countries across Europe, and Slovenia has also been considering the construction of the second unit of nuclear power plant in Krško (hereafter JEK2) for a long time. In Slovenia, due to the

space specifics, it will not be possible to ensure energy independence solely with renewable energy sources (RES Slovenia, 2022). With the abandonment of fossil fuel sources for electricity production and the planned closure of the Šoštanj Thermal Power Plant, JEK2 is becoming an increasingly realistic fact (NSPIP, 2022). The existing Krško Nuclear Power Plant (hereafter NEK) was built in a time of different social and economic conditions and in a completely different legislative framework. Today, however, based on the experience of the past 15 years, the procedures for the placement of arrangements of national importance are very long in our country. The fact that the nuclear power plant is one of demanding facilities also has to be taken into account, from the point of view of coordination with other arrangements in the space, from the technological point of view, and from the point of view of social acceptance. Therefore, the actual time component of the JEK2 realization is a complex issue, which is influenced by many risk factors, such as political decision, legislative procedures, public consensus, supplier selection, project financing, coordination with other arrangements in the area and technological complexity of the project. The key challenge is to reduce all project risks, and to ensure the fastest possible realization of the project, with which we will be able to achieve the set goals regarding the reduction of greenhouse effects, economic development, as well as economic stability.

111

## Geological and Seismological site Investigations For Building New Nuclear Power Plant

Mojca Planinc, Aleš Kelhar

GEN energija d.o.o., Slovenia

mojca.planinc@gen-energija.si

The planning and evaluation of a site for nuclear new build is a critical step in succesfull development of a new nuclear project. Site selection, characterisation and evaluation form an integral part. Comprehensive geological, seismological and geotechnical site evaluation program is usually implemented with the aim to confirm the acceptability of the site and obtaining site-related data. Site investigation activities are organized in cooperation with three main partners: investor, expert scientific team and independent reviewer team.

This article reviews the geological siting aspects and investigation methods. Basically there are two products of the program for assessing seismic hazards at a given site: vibratory ground motion hazards and the potential for tectonic ground deformation (fault capability). Site investigations are performed to accommodate recommendations and requirements and are performed in accordance with international regulatory codes and standards, as well as domestic, national requirements in nuclear industry. General requirements for site investigations are discussed.

It is important to point out the extent of the geological investigations starting with the site and its vicinity, as well as on the near regional and the regional scale. Tectonic and geomorphic analyses and reconnaissance field checking is required to support characterization of seismic sources within the Site Near Region (SNR) and assessment of fault capability within the Site Vicinity (SV). Investigation of geological-tectonical features of the site is needed with particular aim on the assessment of the capability fault at and around the site, characterization of site seismic hazard, and obtaining the design basis earthquake. A brief description of the

methodology of the geological, seismological, geophysical, geotechnical and hydrogeological investigations is provided.

To adequatelly address the terrain geology review of the existing studies and field data is performed, which includes previous field observations, executed field tests, obtained data from laboratory testing and conclusions with analyses and calculations.

The planning of individual site investigations is usually based on the shallow geophysical investigation. Paleoseismic investigations provide information to characterize tectonic history and assess fault capability that are encountered in the site vicinity. Determining an appropriate trench location is a critical part of successfull paleoseismologic studies Information obtained also support updating the seismotectonic model and characterizing seismic sources. Certain conditions must met to identify a suitable location for paleoseismic investigations. Trench sites are selected by integrating different sets of results (geomorphic, HRS geophysical, drilling, shallow geophysics).

Seismic survey gives a unique dataset for understanding the spatial relationship between individual fault segments. The seismic survey and other geophysical data enable understanding of the tectonic evolution of the area and sequence of geological events. Site-scale investigations are aimed on the characterization of local geotechnical and hydrogeological conditions. The geotechnical investigations provide data for the evaluation of site response, assessment of the liquefaction hazard and foundation design. Data obtained with a particular technique compliments other activities to make an informed decision when interpreting results. After obtaining all relevant field data and analysis earthquake engineering analyses takes place. Complimentary skills and collaboration of experts working in different fields of expertize is mandatory for the overall success of the project.

112

## Role of Small Modular Reactors in the Future of Nuclear Power

Tomi Živko, Tomaž Nemec

Slovenian Nuclear Safety Administration (SNSA), Slovenia

tomi.zivko@gov.si

There is a large experience with construction and operation of NPPs with power of 1000 MWe or larger. Accidents at TMI-2 and Chernobyl NPPs strongly affected nuclear industry. One of results was decrease in the number of new builds leading also to reduction of the number of experienced staff. This resulted in more delays at construction sites of new builds due to various problems with planning, supervision, and workmanship, so it increased building costs and risk to an investor.

In recent years, the concept of Small Modular Reactors (SMR) gained popularity. The SMR have lower power (up to 300 MWe) in comparison to NPPs. The SMR are called modular because they are designed for modular manufacturing in factory production and transportation as modular units to a location for installation. Serial production in a factory can improve quality of manufactured systems, lower the cost of single reactor unit, shorten the time of construction and make SMRs economically feasible. Therefore economy of scale is swapped for economy of mass production. The SMR can be used in many different ways such as grid and industrial electricity production, heat cogeneration and desalination. At this moment there are more than seventy SMR designs under development for different applications. Some SMR designs are simplified versions of existing power reactor designs while many other SMR designs are based on new technologies. Proposed SMR designs are generally simpler and often rely on passive systems.

International organizations such as IAEA, OECD/NEA, EUR and WENRA conduct activities connected with SMRs. The IAEA started a Nuclear Harmonization and Standardization Initiative (NHSI), the OECD/NEA prepared The NEA Small Modular Reactor Dashboard which is the SMR overview of the assessed readiness for a particular SMR installation. The EUR developed European Utility Requirements that include light water SMR. The WENRA, the European regulators' association, is also active in challenges related to the development of SMR. Some regulators from different countries have joined their resources and work together in SMR design assessments.

The Slovenian Nuclear Safety Administration (SNSA) is active within the IAEA NHSI. The NHSI goal is to develop a joint approach of different regulators, designers, vendors and operators for the use of SMR technology. The NHSI shall harmonize licencing requirements, safety standards for design and the technology assessment. This is challenging because the licensing legislation, such as requirements on design, safety and staffing is focused on large reactors in some countries and can be too much of a burden for smaller installations.

The SNSA is collaborating in the working group 3 of the NHSI for Leveraging Other Regulatory Reviews.

Currently, there is no initiative in Slovenia for installation of a SMR but it can change in the future. The SNSA needs to raise its readiness to provide timely licensing of such projects. The paper will give an insight into the SNSA activities in the field of SMR as well as a review of international activities.

113

## Nuclear Power Plant Design Assessment

Jurij Kurnik, Robert Bergant

GEN energija d.o.o., Slovenia

jurij.kurnik@gen-energija.si

Nuclear power plants have become an essential source of energy worldwide due to their low carbon emissions and ability to produce large amounts of electricity. However, ensuring the safety and reliability of these facilities is critical, given the potential risks associated with nuclear energy. Therefore, nuclear power plant design assessment is a crucial process that involves evaluating the safety and security of nuclear power plant designs before construction.

This assessment process includes a thorough review of the plant's design, operation, maintenance, and emergency preparedness plans. It involves examining various factors such as the reactor's safety features, containment systems, cooling systems, and emergency shutdown mechanisms, among others. Depending on the purpose of the assessment, the process is carried out by regulatory bodies, technical support organisations (TSOs), and/or other experts to ensure that the proposed design meets safety, regulatory standards.

During the development of JEK2 project (new nuclear power plant build project at Krško site) experts from GEN energija have been actively involved in the activities of EUR (European Utility Requirements) organization and its new nuclear power plant design assessment projects. GEN experts have been involved in last three design assessment, i.e. VVER-TOI, EU-HPR1000 and APR1000 reactors. Furthermore, we have also assessed AP1000 design against EUR key issues and WENRA requirements for existing and new reactors.

GEN has also initiated a request for information process with the vendors of potential reactors for JEK2 project with the intent of acquiring more information on the potential reactors and to enable a direct comparison and assessment of the reactors.

The article provides an overview of various nuclear power plant design assessment processes, assessment scopes and lessons learned from the performed activities.

---

115

## Integration of Heat-only Small Modular Reactor with Thermally Driven Systems

Hussein Abdulkareem Saleh Abushamah[1], Tomáš Kořínek[2], Radek Škoda[1]

[1] University of West Bohemia, Czech Republic

[2] Czech Institute of Informatics, Robotics and Cybernetics, Czech Republic

abushama@fel.zcu.cz

The nuclear small modular reactor (SMR) technologies represent potential competitive carbon-free solutions for replacing fossil fuel-based energy generation. In addition to several SMR technologies designed for electricity generation, some others, like Teplator, are under development for district heating applications. As the heating demand fluctuates over time, using the excess heat during the low demand periods could enhance the load following flexibility, capacity factor, and the economics of the integrated system. Depending on the output temperature of the nuclear plant, the excess heat could be used for different secondary applications. This study aims to evaluate the integration of heat-only reactors with secondary thermally driven applications such as the solar chimney concept and thermoelectric tubes for low-temperature electricity generation and other heat-driven industries like water desalination and hydrogen production. The calculations will be based on a typical heat-demand profile, where the overall efficiency, capacity factor, and secondary products will be estimated for different candidate integrations.

116

# Potential Of Using SMR Reactors At Coal-Fired Power Plant Locations

Jan Lokar, Aleš Kelhar, Robert Bergant

GEN energija d.o.o., Slovenia

jan.lokar@gen-energija.si

Coal is responsible for the largest share of carbon dioxide emissions from the energy sector, making its phase-out key to tackling climate change. On the other hand, the International Panel of Climate Change (IPCC) considered 90 pathways with emission reductions to limit average global warming to less than 1,5 °C. The IPCC found that on average, the pathways for the 1,5 °C would require installed nuclear capacity to reach 1.160 GW by 2050, up from 394 GW in 2020. This growing role for nuclear power will complement variable renewables, which are set to rapidly grow in all climate mitigation pathways. The ambitious target to triple installed nuclear power until 2050 can be achieved with long-term operation of existing plants, building large-scale Generation III/III+ new builds and with deployment of Small Modular Reactors (SMRs).

The development of SMR reactors is dynamic and, in the future, the SMRs can play important role in the low carbon energy policies in the world. Many countries aim to abandon the use of fossil fuels for electrical energy production, particularly coal-fired power plants. The location, infrastructure and employees in coal-fired power plants have the potential for restructuring and deployment of SMR reactors.

In recent years, several studies have analyzed the possibility of implementing SMR reactors at coal-fired thermal power plant locations.

The first part of the article provides a brief overview of SMR development and the recent coal-to-nuclear projects and studies. The second part of the article represents the overview of coal-fired power plant systems and buildings that may be applicable to SMR power plants and discusses potential benefits of coal-to-nuclear projects and challenges to be addressed in the near future.

The last part of the article analyses the options for deploying SMR reactors on the sites of Slovenian coal-fired thermal power plants.

117

# Thermal Energy Storage Combined With a Molten Salt Reactor

Pia Fackovič Volčanjk[1], Andrej Senegačnik[2]

[1] GEN energija d.o.o., Slovenia

[2] University of Ljubljana, Faculty of Mechanical Engineering , Slovenia

pia.fackovic@gen-energija.si

Electricity plays a crucial role in contemporary society and is becoming even more vital with the expansion of electrification. This is causing a worldwide surge in demand for power, the production of which has to be decarbonized to approach the goal of zero greenhouse gas emissions. This can be achieved by increasing the share of low carbon energy sources, particularly renewables and nuclear energy. As the share of variable renewable energy systems, such as solar and wind, continues to rise, the need for power plants, which can adjust their load to match changes in electricity demand and variable supply from renewables, also increases.

Nuclear power plants contribute to maintaining electricity security and power grid stability. They can adjust their operations to a certain extent in response to changes in demand and supply, however they are primarily utilized as base load sources due to their low operational cost. The combination of advanced nuclear reactors and thermal energy storage presents a promising solution for achieving greater flexibility of nuclear power plants, which would allow better alignment with fluctuating demands and supply shifts.

Molten salt nuclear reactors are an advanced form of nuclear power technology. They can operate at high temperatures and low pressure, conditions which result is an increased power plant efficiency and a lower risk of fatal release of volatile radioactive materials. Moreover, the high operating temperature of these reactors makes it capable for integration of high temperature thermal energy storage.

This article describes a concept for thermal energy storage using molten salt combined with a molten salt nuclear reactor that has a thermal power of 750 MW. With this system the nuclear reactor can operate at full power even at times of low power demand and store the surplus heat into the tanks with molten salt for the later use during peak power demand periods. Accumulated heat is used for steam production and extra power in range of 240-500 MW depending on grid demand.

# *Nuclear Regulation, Society and Environment*

201

# Use of Commercial grade items in nuclear safety related applications: New developments and guidance

John Kickhofel[1], Zoran Heruc[2]

[1] Apollo+ Nuclear, Switzerland

[2] Nuklearna elektrarna Krško d.o.o., Slovenija

jkickhofel@apolloplus.com

This paper explains the developments and trends in the European nuclear industry supply chain which have led to an expansion and new European guidance on the subject of commercial-grade dedication (CGD). The practice of verifying critical characteristics of commercial-grade products has expanded to jurisdictions not traditionally connected to the U.S. NRC regulatory space. International organizations including the International Atomic Energy Agency (IAEA), European Commission Joint Research Center (JRC), Nucleareurope, and the International Standardization Organization (ISO) have all been addressing this subject in recent years.

CGD, an acceptance process, which is intended to gain confidence in the quality/conformance of products where were not controlled according to nuclear QA/QC requirements, has a long history of use in some countries' nuclear power programs while other regulatory bodies and licensees have only recently implemented or begun to pilot the process. In Europe, CGD has been supporting procurement of spare parts and replacements in various ways. For example, the smooth transfer of surplus inventory from permanently shut down nuclear power plants to operating plants, saving time and money and even helping to resolve obsolescence issues has been made possible by CGD. The methodology is also foreseen to be beneficial to new reactor deployment in a number of ways.

In 2020, for the first time, European nuclear industry came together to prepare an industry guideline which it intends to follow as a means of harmonizing and ensuring a proven, best practice is adhered to and to provide confidence to regulatory bodies. The Nucleareurope (formerly, FORATOM) Guideline on CGD (full name: Quality Assurance Guideline for Procuring High-Quality Industrial Grade Items Aimed at Supporting Safety Functions in Nuclear Facilities) was authored by Apollo+ in 2021-2022 on behalf of a steering group which included a wide range of European nuclear industry associations, license holders and a vendor. This paper discusses the Nucleareurope Guideline, its development, contents and recommended approach to CGD for license holders and suppliers.

Further accelerating the application of CGD in the supply chain within Europe is an international consensus standard from ISO for nuclear suppliers. That standard, ISO 19443:2018 (Quality management systems — Specific requirements for the application of ISO 9001:2015 by organizations in the supply chain of the nuclear energy sector supplying products and services important to nuclear safety (ITNS)), includes requirements for verifying the critical characteristics of commercial-grade items when they will be used as items important to nuclear safety. Certification to this standard is becoming a requirement license holders are including or intend to soon include in purchase orders for safety-related structures, systems and components as well as services, especially in France and the UK.

The CGD methodology is in fact based on fundamental QA/QC activities familiar to any industrial supply chain. We present a holistic view of the methodology using nomenclature which may be better suited for widespread use across jurisdictions and within the supply chain itself. The CGD methodology enables license holders and manufacturers to use today's high quality commercial standards like those from the American Petroleum Institute (API) The introduction of CGD as a part of the procurement process can support long term operation, O&M cost reductions, and resolving obsolescence cases.

202

## Correlation Between Public Acceptance of Nuclear Technology and Trust in Scientists – Case Study Croatia

Zdravka Škugor Ferdebar[1], Krešimir Trontl[2], Mario Matijević[2]

[1] ROOMOR, Croatia

[2] University of Zagreb Faculty of Electrical Engineering and Computing, Croatia

zdravka.skugor@gmail.com

Public acceptance and support for nuclear technology is one of the key issues affecting potential introduction of nuclear option into energy development strategy and countries' energy mix. There are many factors influencing formation of public opinion on technology matters. One of them is trust in scientists advocating or opposing particular technology.

Long term research of public opinion on nuclear issues in Croatia was recently enriched by the new survey conducted during 2023 and carried out on more than 2000 participants. The survey was partially devoted to the question on trust in scientists and scientific, as well as, expert organizations. Preliminary analysis of the survey results indicate that Croatian public trust in scientists is high with observed positive correlation between trust and acceptance of nuclear technology. Scientific engagement in Covid pandemic management raised some controversies and affected relationship between public and scientific community. To explore possible effects in the field of nuclear technology, results of the current survey on trust in scientists are compared to the results of the pre-Covid survey conducted in 2016.

203

# Advancements in Remote Monitoring of Alpha Emitters Using Alpha Induced Radio-luminescence

Lingteng Kong, Dave Megson-Smith

University of Bristol, United Kingdom

jn19830@bristol.ac.uk

Alpha emitters such as uranium and its decay products, are a significant concern in the nuclear industry due to their potential harm to human health through ingestion, inhalation, or injection. Despite their limited travel distance, often no more than a few centimeters in air, alpha particles can cause considerable damage to DNA if they enter the body. Current detection methodologies necessitate close proximity to these particles, resulting in a time-intensive, laborious process that increases the risk of exposure and contamination for personnel. Furthermore, increments in contamination levels necessitate the use of more stringent personal protective equipment (PPE), while the decontamination or replacement of contaminated hand-held alpha detectors incurs significant costs. There is an urgent need for devices capable of detecting alpha particles from distances of several meters to enhance safety and efficiency in applications such as glovebox characterization for decommissioning, lab or wide-area contamination assessment, real-time alpha spillage monitoring, and open environment dispersion monitoring.

The basis of our research revolves around exploiting alpha-induced radio-luminescence (RL), a phenomenon where alpha particles excite nitrogen molecules in the air to emit photons. These emitted photons can be harnessed to determine the position of alpha emitters at extended distances. Utilizing a deep-cooled CCD camera coupled with a lens system, we were successful in detecting a 3 MBq point Am-241 alpha source from a distance of 30 meters within a span of five minutes. The resulting high-resolution image enabled us to locate alpha sources with millimeter accuracy.

One of the primary challenges faced is the fact that alpha fluorescence primarily occurs in the UV region, where the ambient light is billions of times stronger than the alpha fluorescence signal. To mitigate this, we employed a filter system to block the ambient light and conducted tests under different light conditions. Currently, our system is operational under light conditions with minimal deep UV light, such as LED light, inside a glovebox, or at nighttime. Given that sunlight below 280 nm is blocked by the Earth's atmosphere, we are investigating the feasibility of detecting alpha emitters under sunlight with our next-generation prototype.

Our research underscores the potential of this technology in significantly transforming remote alpha monitoring, nuclear forensics, and contamination control during decommissioning. By identifying the photons emitted by excited nitrogen molecules, we can locate alpha emitters without direct contact, thus diminishing the risks inherent in traditional detection methods. The findings from our study exhibit the practicality of this technique and its promise for comprehensive adoption in nuclear industries, offering a safer, more efficient approach for the detection and surveillance of alpha emitters.

204

# Application of Espoo Convention on environmental impact assessment in transboudary context on Extension of life-time for nuclear power plant Krško

Vesna Kolar Planinšič

Ministry for Environment, Climate Change and Energy, Slovenia

vesna.kolar-planinsic@gov.si

This paper discusses the national, European and international aspects of environmental impact assessment (EIA) with relation to periodic safety review and with respect to the lifetime extension of nuclear power plants. In an era of tackling climate change and searching for low-carbon sources of energy, when many countries are deciding to close old nuclear power plants, while those which have continuously invested in their improvements are convinced that they can ensure their safety and reduce their environmental impacts are extending their operational lifetime, this is an open and burning issue. The paper presents the content of environmental and safety impact assessment and legal and practical application. It llustrates the application of the Convention on Environmental Impact Assessment in a Transboundary Context and Directive 2011/92/EU of the European Parliament and of the Council of 13 December 2011 on the assessment of the effects of certain public and private projects on the environment, as well as Slovenian Environmental Law as a legal basis for transboundary consultation on likelyhood of environmental impacts and their assessments. The paper discusses the main international case of Rivne NPP from Ukraine and the case law of the European Court of Justice (ECJ) with respect to the nuclear power stations Doel 1 and Doel 2, from Belgium, which represent important legal practice and they developed in paralel with the EIA Screening in Slovenia.

Furtheron the application of new practice in transboundary environmental impact assessment for the lifetime extension in the first EU case of the Krško Nuclear Power Plant is presented. The effective transboudary consultation include technical consultation on EIA documentation, translation of documentation in official languages of affected Parties, quality control of documentation, paralel public consultations in all countries and answering to all questions of the potentially affected Parties.

The main legal steps were discussed in advance within UNECE/Espoo focal points as follows: notification, preparation of documentation, assurance of EIA quality, consultation with ministries and organisations, consultation with public, preparation of final decision, translation of final decision, informing on final decision. The application methods for public consultations in Austria were open public presentations, followed with national television. The transboudary technical consultations were implemented with five EU countries, all Parties to Espoo Convention: with Croatia, Austria, Germany, Hungary and Italy. Comments on safety, environment and health, waters, waste storage and mitigation measures and others were discussed on the expert level between two teams. In all countries the expert teams were established and technical consultation organised also on location with the presentation of the project Extension of life time of NPP Krško for 20 years and presentation of existing NPP. In addition the technical team present the project and EIA in public hearing in Austria and Croatia. All comments, and opinions from Parties, and of Slovenian Nuclear Safety Administration; Ministry for public health, Slovenian Water Agency, Institute for Nature Conservation and NGO were concidered and taken into account to the extent appropriate and the environmental protection consent was issued. All procedure was organised in efficient way in1,5 years, from october 2021 until january 2023.On the base of the criteria: timing, involving public consultation, active

technical international cooperation with five EU countries, paralel consultations with expert teams, no complain at the court, Krško NPP EIA transboudary procedure could be considered as a good practice in transboudary context.

205

## Evaluation of the Justification for Using Consumer Products and Geological Samples Containing Radioactive Substances

Tinkara Bučar, Matjaž Stepišnik, Matjaž Koželj

Institut "Jožef Stefan", Slovenia

tinkara.bucar@ijs.si

A strategy was developed at the Jožef Stefan Institute to evaluate the justification for using consumer products containing radioactive substances and geological samples (rocks and minerals) that contain naturally occurring radioisotopes. This project was funded by the Nuclear Safety Administration (SNSA). Some commercially available products can contain radioactive material, and they are freely available to individuals without special control by administrative authorities; such products are referred to as consumer products. It is necessary for the authority to evaluate whether there is a valid justification for their continued use when these products are on the market or if they are already in use.

The project aimed to establish acceptance criteria for justifying the continued use of consumer products, to warn the public of potential risks, and to publish recommended methods for handling such objects. The main goal was to enable residents to collect and handle these objects safely.

According to Slovenian legislation, a radiation protection assessment is necessary, and it must be ensured that an individual from the population will not receive an effective dose greater than 10 µSv per year. The annual effective dose for geological samples with natural radionuclides of 1 mSv is considered as a criterion. The equivalent skin dose is not defined in the legislation for these cases, so a value of 0.5 mSv was used for consumer products, and 50 mSv natural radionuclides in geological samples. These values were adopted as generic criteria for received annual doses.

The received doses for different groups of consumer products and geological samples using different exposure pathways for normal use and accident scenarios were calculated. The dose criterion cannot be used effectively for quick decision-making. This is the reason for developing operational intervention activity levels (AOIL) at which the estimated annual dose would reach the generic criterion. The predicted dose rate and the response of the contamination probe were estimated using the AOIL.

A summary of calculations and assessments in short reminders was prepared for authorized experts and administrative authorities. The reminders include a description of the consumer products, radiological data, the intended method of use, a statement on the justification of use, conditions of use, and termination of use. They also contain examples of typical activities and dose rates for consumer products and estimated operational intervention levels (OILs), including activity, dose rate, and the response of the contamination probes. Additionally, the scenarios used and the equations for the calculations, along with the parameter values for individual groups, were added.

206

# Implementation and application of a GitLab CI-pipeline in support of V&V activities of AC2

Vincenzo Anthony Di Nora, Andreas Wielenberg, Thorsten Hollands

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Germany

Vincenzo.diNora@grs.de

System codes are employed to support safety analyses and safety demonstrations of nuclear facilities with evidence tools that use best-estimate models. As evidence tools that support a safety case or its assessment by a regulatory authority, these codes must be properly qualified in line with regulatory requirements. This includes a systematic and suitable verification and validation (V&V) process during the development of these codes.

The AC2 code package, developed by GRS to analyze operational conditions, design basis and design extension conditions up to severe accidents in nuclear facilities, applies a V&V process in line with current regulatory expectations. Recently, the AC2 development process is supported and documented using the GitLab® platform run on a server operated by GRS. This allows one to make use of GitLab's automated testing and continuous integration (CI) features, which have been applied to perform basic unit tests of the code as well as verification calculations. In parallel, regression tests are also performed on a Jenkins® server. In addition, comprehensive V&V activities are continuously conducted by a dedicated validation team outside of GitLab.

The CI capabilities of GitLab can be used to further improve the V&V process for AC2 and support the V&V experts in their daily work. One effective solution is to implement automated and well-structured CI test trains, known as pipelines in GitLab. This work introduces the respective activities for setting up a CI-pipeline under GitLab to run a well-balanced suite of test cases for the V&V of the entire AC2 package. Starting from GRS's in-house plot tools and basic CI pipeline configurations, the development focused on designing and implementing a tool that can: Flexibly retrieve already built executables of the package for testing, and/or build executables for development branches as specified by the pipeline user; using the selected executables to perform calculations on a verification or validation matrix, which can also be configured by the user; comparing the solutions for the different code versions by automatically generating a set of figures of merit where also measurement data for the tests cases can be displayed, and finally, automatically produce a well-structured report file for the user, with which the assessment of the code versions on V&V can be performed.

The newly implemented CI pipeline has been tested extensively and has been successfully applied to assist the V&V activities of the validation team in connection with the release process for AC2-2023 planned later in 2023. For demonstration purposes, this paper presents and discusses the application of the pipeline on two tests: One involving a case without core degradation scenarios and another involving a case that does consider core degradation. The flexibility and intuitive implementation of the pipeline make it a valuable and efficient tool for the V&V of AC2, and potentially for the V&V of other simulation codes for nuclear applications as well. As the tool is still under development, options and plans for further improvements are discussed.

207

# Evolutionary computing for the estimation of radiation fields

Sebastian Thomas Oakes

University of Bristol, United Kingdom

sebastian.oakes@bristol.ac.uk

The tracking radiation levels in the surrounding environment is of utmost importance to a number of scientific fields, whether it is regular monitoring of nuclear facilities during reactor operation, or the need for accurate, timely maps of radiation levels during disaster response scenarios.
Evolutionary computing (EC) methods take inspiration from the natural process of evolution in the biosphere, in which limited resources force individual agents through the filter of natural selection. Those agents best adapted to a given environment will survive more frequently, passing on combinations of genetic material to the next generation. This natural iteration process gradually shifts the overall population towards better 'fitness'.

EC takes the main components of this process (evaluation, selection, recombination and mutation) to solve computational problems. In the case of radiation mapping, the problem is one of optimisation - the radiation measurements at certain positions are known (output), the model of how radiation interacts with matter is understood (the model), but the exact ditribution of emitters in the surroundings is unknown (input).

Exploration of the whole solution space through brute force becomes unfeasible due to the high number of variables at play, but a guided exploration of this space can yield useful solutions. By defining what a well adapted solution looks like, we add selection pressure to a population of potential solutions.

This work explores the feasibility of applying EC to the mapping of estimated radiation levels in the surroundings. Different potential solutions (emitter distributions) are generated for the population, before being evaluated for fitness, by simulating the effects of the solution emitter distribution on the detectors. The best candidates are then recombined and mutated to populate the next generation. This process iterates to effectively explore the solution space, guided by the predefined selection pressure.

A simple 2-dimensional simulation environment is first created to assess the viability of the method, and explore any issues that arise that are specific to radiation measurement/simulation. The method is then extended to 3 dimensions, using robotically gathered lidar and gamma spectrometer data as the inputs and output respectively.

# Changes in the field of Occupational Safety and Health (OSH) at Krško Nuclear Power Plant

Ivo Žarković, Rudi Janežič

Nuklearna elektrarna Krško d.o.o., Slovenija

rudi.janezic@nek.si

At the Krško Nuclear Power Plant (NEK), we are committed to ensuring occupational safety and health (OSH) in accordance with the Occupational Safety and Health Act and to the highest standards and best industrial practices in the world.

The marked increase in the number of activities due to intensive work on the *Safety Upgrade Project* and a larger number of contractors with lower OSH standards caused a deterioration in the OSH performance. We recorded a greater number of work injuries, near misses, and cases of inappropriate work practices.
In April 2019, the independent expert review mission in NEK identified an *area for improvement* (AFI) in the OSH field.

We analysed the situation and identified the root causes and contributing factors:

- Human behaviour, lower OSH standards in the work of contractors, and a lower level of contractors' competence in the OSH field.
- Low awareness of the importance of the OSH field.
- NEK's rules and expectations are not always clear and unambiguously understandable.
- Managers do not sufficiently highlight the expectations in the OSH field and do not insist on consistent compliance with rules and standards.
- The *Corrective Action Program* is not being used effectively to address human misconduct in the OSH field.

We have identified and determined the riskiest areas:

- work at height,
- lifting and transferring loads,
- work in confined spaces, and
- risk of electric shock.

Based on these findings, we defined actions to improve the situation:

- Introduction of the campaign *I work safely* on the importance of OSH and safe work with the help of posters, an internal information system (Plant Portal and Screen), and information on the OSH events.
- Introduction of a counter of days without injuries at work.
- Construction of an OSH Hands-On Training Centre and establishment of an OSH Hands-On Training Program for NEK employees and contractors.
- Clear definition of NEK's requirements and expectations in the OSH area.
- Regular inspections, tours of construction sites and demanding workplaces at NEK from the point of view of work safety, fire safety, and compliance with the NEK's rules and requirements.

- An expert visit to a nuclear power plant that achieves exemplary results in this field.

After re-examining the OSH effectiveness in October 2021, we found that our efforts to carry out the work carefully and safely did not bring adequate results. Therefore, the management of the power plant demanded that we all strongly, responsibly, and immediately ensure compliance with the OSH principles. We decided that we need to change the approach to safety and health by increasing the responsibilities of NEK managers, changing the mindset of all NEK employees and contractors, and thus improving the safety culture at NEK.

In the *NEK's Commitments and Goals* for the years 2020, 2021, 2022, and 2023, the NEK's Management Board determined that one of the priority areas was to *consistently observe the principles of occupational safety and health – always and everywhere*.
We have taken additional measures:

- Introduction, promotion, and observance of the *Safe Work or No Work* principle, especially for risky areas.
- Aggravating the responsibility of all NEK managers.
- Aggravating the responsibility of contractors and determining contractual penalties.
- Mandatory free hands-on training for all risk areas, including for all contractors.
- Stopping work in the event of identified deviations in compliance with the OSH measures.
- Mandatory participation of project managers in inspections of construction sites/workplaces in the area of OSH.
- Preparation of additional OSH performance indicators.
- Hiring additional OSH professionals – specialists for safe work at heights.
- Creation of an application for keeping OSH minutes.
- Change in the operation of the NEK's OSH professionals.

The *I Work Safely* campaign is ongoing. The purpose of the campaign is to warn and raise awareness of the importance of safety and health at work. Safety messages are printed on posters throughout the plant. Content from this area is included in the *Weekly Safety Consideration*, we play short video films on the information screens and publish important OSH messages.

In the new OSH Hands-On Training Centre, we have established hands-on training sites for all four identified risk areas. Everyone who faces certain risks in any way must undergo training. We regularly train NEK workers and all contract workers. We have also effectively extended hands-on training to the working environment, where the NEK's OSH experts actively help workers to secure the workplace, encourage and direct them, and ensure the safe execution of work through supervision.

We regularly monitor workplaces and construction sites with inspections attended by all persons responsible for the safe implementation of projects. The results of the inspections are minutes, in which all deviations and measures to eliminate deficiencies are recorded. We have developed an application for keeping minutes, which we use for the newly introduced OSH performance indicators.

All our measures are regular and permanent; the measured results confirm that we have made great progress. We have greatly increased the awareness, mindset, and safety culture of workers. We have changed the behaviour of workers; we ensure the safe execution of all works at NEK.

# On the applicability of the IAEA documentation to innovative reactors

Andreas Ikonomopoulos[1], Spyros Andronopoulos[1], Mirela Nitoi[2], Walter Klein-Heßling[3], Danilo Ferretto[4], Alain Flores y Flores[4], Mitja Uršič[5], Rok Krpan[5], Boštjan Zajec[5], Leon Cizelj[5], Timo Löher[3], Gian-Luigi Fiorini[6], Egidijus Urbonavičius[7], Kateryna Fuzik[8], Mykola Sapon[8], Ismo Karppinen[9], Francesco Lodi[10]

[1] National Centre for Scientific Research "Demokritos", Greece;

[2] Institute for Nuclear Research (RATEN ICN), Romania

[3] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Germany

[4] National Radiation Protection Institute (SURO), Czech Republic

[5] Institut "Jožef Stefan", Slovenia

[6] LISTO BVBA, Belgium

[7] Lithuanian Energy Institute, Lithuania

[8] State Scientific and Technical Center for Nuclear and Radiation Safety (SSTC NRS), Ukraine

[9] VTT Technical Research Centre of Finland Ltd, Finland

[10] Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA), Italy

anikon@ipta.demokritos.gr

There is a need to "… facilitate the establishment of a common understanding on licensing methodologies for advanced technologies between nuclear safety regulators, contributing to harmonisation of licensing methods of future installations."[1] It is the enhanced interest in designing and constructing novel nuclear reactor concepts – collectively identified as 'evolutionary and innovative designs' (EIDs) – that has spurred the requirement to examine the current IAEA safety standards under the prism of their appropriateness to innovative reactor designs while also recognizing potential gaps that may exist.[2] On that matter, IAEA has performed comprehensive reviews on the applicability of a number of safety standards to various types of non-water cooled reactors and small modular reactors (SMRs).

In this framework, the HARMONISE [3] project has performed a high level review of the applicability of IAEA Safety Standards to SMRs and Generation IV reactor concepts identified by the Gen-IV International Forum (GIF) in the GIF Technology Roadmap [4], namely: gas-cooled fast reactor; lead-cooled fast reactor; molten salt reactor; sodium-cooled fast reactor; supercritical-water-cooled reactor; very-high-temperature reactor. The EIDs selected for the study have been designed with various types of coolant, fuel, neutron spectrum, inherent safety features and modular concepts. A number of research and development projects worldwide are focusing on these EIDs having achieved various maturity levels.

The evaluation performed covered several among the high-level IAEA documents from the IAEA Safety Standards, i.e., General Safety Requirements (GSRs), Specific Safety Requirements (SSRs) and Specific Safety Guides (SSGs). The works addressed issues related to safety without examining nuclear security, while also considering the findings of similar studies performed by IAEA, WENRA, GIF and the SMR Regulators' Forum as well as the outcomes of pertinent Euratom-funded projects. Despite their generic nature, in particular for GSRs and SSRs, the publications examined are not completely technology neutral, which motivates the recommendations that are formulated in order to make them applicable to all facilities and activities.

The review considered crucial technical aspects such as multi-module SMR concepts, factory-built and potentially factory-fueled designs that ought to be transported as well as cogeneration. Functional and technical topics contemplated include the employment of passive systems, consideration of inherent safety in the implementation of the defence-in-depth concept, deliberation on the term ‘core melt’, recognition of the consequences of low level operating experience, along with facility commissioning, operation and decommissioning. The work was supplemented by gap analyses and subsequent proposals of new and amended requirements. The recommendations emerging from the HARMONISE findings should be considered as proposals put forward for further perusing.

ACKNOWLEDGEMENT: This project has received funding from the European Commission – Euratom under Grant Agreement No 101061643. Views and opinions expressed are however those of the authors only and do not necessarily reflect those of the European Union or European Commission – Euratom. Neither the European Union nor the granting authority can be held responsible for them.

[1] https://ec.europa.eu/info/funding-tenders/opportunities/portal/screen/opportunities/topic-details/horizon-euratom-2021-nrt-01-06, retrieved on 29 May 2023
[2] IAEA, Safety Report on Applicability of Safety Standards to Non-Water-Cooled Reactors and Small Modular Reactors, IAEA Safety Reports Series No.123 [IAEA Preprint] (2022), https://preprint.iaea.org/search.aspx?orig_q=RN:53077569
[3] https://harmonise-project.eu/, retrieved on 29 May 2023
[4] Gen-IV International Forum (GIF): Technology Roadmap Update for Generation IV Nuclear Energy Systems, OECD/NEA, January 2014

210

## Transport of Radioactive Material: “The Road to Success – is Always under Construction”

Janez Češarek

Institut "Jožef Stefan", Slovenia

janez.cesarek@gov.si

The Slovenian stakeholders have rich national and international experiences with different subsets of safe and secure transport of radioactive material, including fissile material. No traffic accidents or serious non-compliances have occurred during the transport of radioactive material in Slovenia in the past years. In the previous decade, the Slovenian Nuclear Safety Administration (SNSA) established an informal group on safe transport of radioactive material – bringing together a dozen of experts from different governmental entities as well as some well-recognised carriers and other counterparts. During the COVID-19 pandemic and afterwards, further activities have been pursued in order to nurture outreach, underlining the importance of the issue (e.g. a dedicated seminar), tailor-made presentations to a few other interested users (consignors/consignees) of radioactive material, etc.

The legislative pillar is undoubtedly the Agreement concerning the International Carriage of Dangerous Goods by Road (ADR) as well as the requirements from the other modal regulation. Nationally, also the Act on Transport of Dangerous Goods ("ZPNB") and the Ionising Radiation Protection and Nuclear Safety Act ("Nuclear Act, ZVISJV-1") shall be tightly followed, together with all the pertinent decrees and rules stemming from them.

The international review cycle of the most important document of the International Atomic Energy Agency (IAEA) in this regard, i.e. the 2018 Edition of the Regulations for the Safe Transport of Radioactive Material ("SSR-6"), has started with many proposals which have been collected and assessed (some of them also earmarked as denied).The main Slovenian stakeholders will also more actively follow the path towards this, so much-needed an update. The indicated changes may not be "tectonic" this time, but multi-faceted anyway, e.g. A1/A2 values, consignment vs. package, special form radioactive material – ageing mechanisms, mixed packing of LSA and SCO in the same package, etc. Later on, the content of SSR-6 will be meticulously streamed into one of the future revisions of ADR – thus obligatory for all of us.

Drivers of radioactive material (dangerous goods, "Class 7"), their organisations and dangerous goods safety advisers (DGSAs) play an important role in the whole chain of transport activities. Periodic trainings of drivers as well as DGSAs are enshrined in ADR. The article will also try to wrap some practical experiences and feedbacks from those trainings – with the aim at further fostering a solid platform – where safety culture (and security culture, too) can be underpinned – to help mitigate and minimise the risks, discuss lessons-learnt from incidents (which have occurred e.g. during the last two decades abroad) and novelties in the legislation that need to be adequately incorporated in the duties to prevent non-compliances.

SNSA has been fairly active within the European Association of Competent Authorities for safe transport of radioactive material (EACA). This non-formal group with abundant expertise of its experts has a frank dialogue and exchange of information on different transport-related issues, sharing good practices regionally and being at the same time a bridge to the IAEA-led endeavours in this sphere. The last-year's IAEA IRRS mission in Slovenia did not unveil any spectacular pitfalls or mediocre performances considering domestic transport of radioactive material. However, such "samplings" and a string of specific questions may also resonate after such a mission – with a subtle yet complacency-contesting question: "What else could be done with present resources and being better and innovative also tomorrow, interweaving resilience and sustainability in transport-related activities which are there – on our roads – on a daily basis". For sure, small and incremental steps are always possible and one the most important things is a practical and vivid dialogue (including inspection control) – with care and exactness with those hauliers that transport high-activity sealed sources or fissile material.

211

## Design innovations and novel safety claims impacting power plant licensing

Francesco Lodi[1], Giacomo Grasso[1], Mirela Nitoi[2], Marin Constantin[2], Minodora Apostol[2], Marco Caramello[3], Alain Flores y Flores[4], Antonio Dambrosio[4], Guido Mazzini[4], Oleksandr Kukhotskyi[5], Anatolii Shyshuta[5], Mitja Uršič[6], Andrej Prošek[6], Boštjan Zajec[6], Spyros Andronopoulos[7], Andreas Ikonomopoulos[7]

[1] Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA), Italy

[2] Institute for Nuclear Research (RATEN ICN), Romania

[3] Ansaldo Nucleare SpA (ANN), Italy

[4] National Radiation Protection Institute (SURO), Czech Republic

[5] State Scientific and Technical Center for Nuclear and Radiation Safety (SSTC NRS), Ukraine

[6] Institut "Jožef Stefan", Slovenia

[7] National Centre for Scientific Research "Demokritos", Greece

anikon@ipta.demokritos.gr

The licensing process of novel reactor installations, based either on fission or fusion technologies, is expected to pose new challenges on the current licensing and authorization processes. The first step for building a more homogeneous and comprehensive framework is to identify the needs for licensing innovative nuclear installations which is one of the HARMONISE project [1] objectives.

To this end, a review will be performed on the specificities associated with innovative reactor technologies in order to point out not only the associated safety-related needs but also their relevance in terms of the licensability of power plants harnessing such novel implementations.

The review will be performed by means of a questionnaire aiming at capturing specific design features of innovative technologies that are anticipated to pose novel challenges from a safety and thus, licensing perspective. The survey intends to engage major stakeholders in European fission and fusion programs with the aim to cover as many technologies as possible.

Complementary to the survey, the works will be extended to cover two reference concepts selected as representative of the fission and fusion technologies: the Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED) and the DEMOnstrator Power Plant (DEMO), respectively. The intention is to extend the review to more detailed levels of the hierarchical framework defining the licensing context (i.e., from general principles to specific requirements).

The questionnaire application to the ALFRED and DEMO concepts will help identifying the role of innovation and its consequences in the safety demonstration of new technologies. These findings will be further utilized to recognize the gaps present in the current licensing frameworks and make recommendations on how to fill them.

ACKNOWLEDGEMENT: This project has received funding from the European Commission – Euratom under Grant Agreement No 101061643. Views and opinions expressed are however those of the authors only and do not necessarily reflect those of the European Union or European Commission – Euratom. Neither the European Union nor the granting authority can be held responsible for them.

[1] https://harmonise-project.eu/, retrieved on 1 June 2023

# *Nuclear Materials*

301

# GANEX gamma radiation tolerance methodology

Chris Hutson

University of Bristol, United Kingdom

chris.hutson@bristol.ac.uk

A new nuclear fuel cycle is being considered as part of the UK's Advanced Fuel Cycle Programme which seeks to enable reprocessing of fast reactor fuel. Current industrial scale technologies for reprocessing nuclear fuel are centred around aqueous separations using aqueous and organic phases to selectively separate and purify various components of the fuel for reuse. An alternative that is in development is pyrochemical processing, which uses molten salt (e.g. a lithium chloride – potassium chloride eutectic) as a solvent for the fuel and electrochemistry to separate the reusable parts from the waste. In the UK, operational commercial-scale reprocessing plants have been mainly operated based on an aqueous reprocessing process called PUREX. The PUREX process uses tributylphosphate to selectively isolate uranium and plutonium from dissolved fuel in the Thorp and Magnox plants at Sellafield. This technology works very effectively for the recycling of uranium and plutonium but has proved wholly ineffective at isolating the minor actinides. If the minor actinides can be isolated and used as fuel then the heat burden and, therefore, size of any future repository can be considerably decreased. Therefore, advanced reprocessing fuel cycles (advanced aqueous and pyrochemical processing) make use of novel organic ligands to isolate the actinides for ultimate continued disposition in nuclear reactors, which fast reactors are especially well suited for. This project aims to establish techniques and capabilities for measuring the tolerance of such organic ligands to high levels of radiation using a sequence of radiation exposures and analysis techniques. This is necessary as fast reactor fuel is generally considered to be much more radioactive than thermal reactor fuel due to its higher initial plutonium/minor actinide content, higher burn-up and shorter cooling times. This additional radioactivity is expected to place a greater burden on radiation tolerance of reprocessing technologies for fast rector fuel relative to light water reactor fuel.

This project has studied radiation effects on the current European reference advanced reprocessing process, called GANEX (Group ActiNide EXtraction). This process has been proven on surrogate feed solutions and spent fast reactor fuel to produce excellent recoveries of the actinides, so this option is being pursued to improve reprocessing, enabling a closed fuel cycle. TODGA is the preferred ligand for solvent extraction as it has excellent properties for minor actinide separation, but has a low capacity for Pu. Therefore in the organic phase, DMDOHEMA was introduced and a ratio of 0.2 mol/L TODGA and 0.5 mol/L DMDOHEMA. Other species are also being considered to improve the loading capacity, physical properties and ease of backwashing extracted metal ions. Recognising that there are several species being considered for the EURO-GANEX and Advanced Fuel Cycle Programme, this project has been seeking to establish methodology for radiation tolerance testing, and investigated two specific ligands which were:

- N,N,N′,N′-Tetraoctyl Diglycolamide (TODGA).
- N,N′-Dimethyl,N,N′-dioctylhexylethoxymalonamide (DMDOHEMA).

The work described herein involves simulation of the intense radiation fields caused by the radionuclides in spent fuel gamma-emission and evaluation of the effect on the GANEX extractant system. This was achieved by exposing the ligands of interest to a high-activity caesium-137 source.

The aim of the work was to begin to fill in current knowledge gaps about radiation effects on these organic ligands. Therefore, following exposure samples were chemically analysed to understand the breakdown products formed by radiolysis. The primary research question being answered is: quantify extent of loss, formation of H2 and formation of degradation products as a function of dose.

302

## Probabilistic assessment of induced intergranular stresses in polycrystalline materials

Timon Mede, Samir El Shawish

Institut "Jožef Stefan", Slovenia

timon.mede@ijs.si

Predicting various ageing mechanisms and material-degradation modes in polycrystalline materials under macroscopic mechanical loading requires the knowledge of stresses induced between crystal grains. These stresses depend on the external loading, material properties and the exact configuration of all the grains in the aggregate (including their sizes, morphologies and lattice orientations). A simple perturbative model has been developed in which each grain boundary is characterized by its orientation and the crystallographic orientations of surrounding grains, while the more distant neighbourhood is modelled by homogeneous and isotropic material. Such setting allows us to solve the constitutive (elastic) equations analytically, while the effect of anisotropic neighbourhood in realistic cases is treated by adding Gaussian fluctuations to the local stresses predicted by the model. Comparison with numerical finite element simulation results demonstrates that while we cannot accurately estimate the induced (local) stresses on individual grain boundaries, we can very well reproduce their distributions on the grain boundaries of a chosen type corresponding to the same grain-boundary strength. That means that while the model is not able to predict the actual crack-initiation sites within the aggregate, it could still be used to compute the probability for cracking.

303

## Stress corrosion cracking behavior of stainless steel 310S in supercritical water

Daniela Marušáková[1], Monika Šípová[1], Michal Novák[2], Radek Novotný[2]

[1] Research Centre Řež, Czech Republic

[2] Joint Research Centre Petten, Netherlands

daniela.marusakova@cvrez.cz

Recent interest in nuclear power technologies is directed to small modular reactors (SMR). The possible and believed option is the one cooled by supercritical water (SCW-SMR). The main objectives of the project ECC-SMART, which is based on SCW technology, are to define the design requirements for the future SCW-SMR, to develop the pre-licensing study and guidelines for the demonstration of safety in the further development stages of the SCW-SMR concept. Based on the scientific results, the stainless steel 310S and other alloys have been selected as the most promising commercially produced materials for application as the cladding material in SCW-SMR. Slow strain rate tests in combination with scanning electron microscopy were used to evaluate the stress corrosion cracking susceptibility in SCW.

304

## Development Progress of Key Materials for Lead-Cooled Fast Reactor

Xinyu Wang

International Academy of Neutron Science, China;

Shandong Key Laboratory of Neutron Science and Technology, China

xinyu.wang@fds.org.cn

Lead Bismuth Eutectic (LBE) has been proposed as the candidate coolant material for Lead-based Reactor, and the high temperature LBE environment creates requirements for the corrosion resistance of structural materials. As a potential candidate for next-generation structural materials for advanced energy systems, the FeCrAl alloy stimulates broad research interests because of its superior high-temperature corrosion resistance, which could be attributed to its surficial oxide layer, Al2O3. A protective oxide film is the key to the corrosion resistance of the FeCrAl alloy. The mechanism of the formation of the multilayer oxide film of the FeCrAl alloy in 700 °C air was explored by studying the structure evolution of the oxide film and the oxidation kinetics of FeCrAl. The results show that a multilayer oxide layer is formed on the surface of the FeCrAl alloy after 1344 h, with a (Fe,Cr)2O3 layer, an Al-rich oxide layer, an Al-depleted zone, and a new Al-rich oxide layer sequentially arranged from the surface to the matrix. This indicates that the Al element plays an important role in the formation of the oxide film. The Al in the matrix is depleted to form the Al-rich oxide layer, resulting in the Al-depleted zone. The new Al-rich oxide layer is formed under the Al-depleted zone by internal oxidation. It should be noted that the precipitation of the AlN phase in the matrix is observed, which might be a probable factor for the Al-depleted zone in the matrix.

305

## Characterisation of Graphite using Raman Spectroscopy

Jarred Minards

University of Bristol, United Kingdom

rn20062@bristol.ac.uk

This research focuses on the use of stand-off Raman Spectroscopy to characterize moderator graphite in-situ.

There is a great need for more tools to assist in graphite characterisation, in the UK Magnox fleet of reactors the approach to decommissioning the cores of graphite-moderated reactors is still uncertain. The most significant challenge associated with removing graphite from the nuclear sites is understanding the distribution of carbon-14 in the graphite. To assist in this effort the use of in-situ Raman spectroscopy can help identify surface deposits that consist of carbonaceous compounds and result from the radiolytic oxidation of the graphite core during operation. The use of stand-off Raman to identify these surface deposits circumvents the need to trepan the graphite fuel channels and transport the material to a laboratory for analysis. This significantly reduces the time needed to characterise the graphite and does not need equipment for trepanning to provide the same analysis.

Raman Spectroscopy has already been used to characterise active graphite from moderator cores. This PhD project plans to use the knowledge gained from academia and the development of a new sensor package system to deploy and map out a fuel channel in a reactor graphite core. This sensor package will provide information about surface deposits in the fuel channel and also provide information on the condition of the surface of the fuel channel without the need to trepan or transport active graphite off of a nuclear site.

306

## Prospecting of lithium deposits through aerial mapping of naturally occurring radioactive material (NORM) using CsI detectors

Ewan James Woodbridge[1], Dean Connor[1,2], Yannick Verbelen[1], Peter Martin[1], Thomas Bligh Scott[1]

[1] University of Bristol, United Kingdom

[2] National Nuclear Laboratory, Warrington, United Kingdom

ewan.woodbridge.2018@bristol.ac.uk

Gamma radiation mapping is used for a variety of applications, from radioactive source localisation to geological mapping. Different rock lithologies contain different amounts of naturally occurring radioactive material (NORM), allowing the radiation levels present to indicate possible changes in rock types. Uncrewed aerial vehicles (UAVs) have been used as a method for gamma radiation surveys for the past two decades, as it offers the ability to radiometrically survey the area without exposing the pilot directly to the area of interest as well as enhanced areal coverage compared to those by ground-based means. NORM radionuclides such as 40K, 238U and 232Th exhibit higher gamma energies than those of anthropogenic origin such as 137Cs and 60Co but are often found in much smaller concentrations in the environment, and as a result, are more difficult to detect under normal UAV measurement conditions. Scintillator detectors are routinely favoured for NORM mapping due to being cheaper to develop into larger crystal volumes than those of semi-conductor detectors, resulting in a greater sensitivity to the higher energies because of increased probability of complete absorption of high energy photons in the sensitive volume.

This study compares two different volumes of CsI scintillator detectors (36cm3 and 237cm3) for NORM mapping by undertaking a field study and comparing these to GEANT4 simulations. The in-field study was completed in a lithium-bearing granite open pit quarry in Cornwall where the 40K concentration was mapped by both detectors mounted to a UAV. Despite the higher cost associated with the 237cm3 detector, the larger volume detector shows a considerable increase in the resolution of the gamma spectra produced by the NORM in the pit and found to be more suitable for NORM mapping than the 36cm3 detector. It was concluded that there is a correlation between the radioactivity observed and the presence of a potassium signature. Where the lithium-bearing granite resides there was a decrease in radioactivity and subsequent potassium signature and so it may be possible to map the occurrence of lithium bearing ores by changes in the observed radioactivity.

---

307

## Enabling accurate characterisation of thermal properties in nuclear materials with an open-source software suite PULsE

Olga Vilkhivskaya[1], Artem Lunev[2]

[1] RAOTEH (subdivision in Gatchina), Russian Federation

[2] Independent Researcher

vilhivskaya@raoteh.ru

Light/laser flash analysis instruments have been used extensively over the past years to infer the thermal properties in fusion- and fission-related nuclear materials. Example applications include fresh and irradiated zirconium alloys, nuclear fuel, tungsten alloys, and others. Experimentally, only a voltage-time curve is typically registered, where the voltage is proportional to the sample heating. For this data to be useful, it must be processed with an inverse-problem solver of a relevant heat transfer problem. Normally this is done via commercial software, e.g., by Netzsch, Linseis, TA Instruments, etc. As such, PULsE [1] is the only currently available open-source analysis package for LFA experiments able to read datasets from different instruments and do the processing. The advantages of using PULsE for LFA data processing include a correctly implemented numerical pulse correction, high-accuracy solvers for opaque [2], transparent [3] and translucent [4] samples, optimisers robust to various kinds of noise (e.g., Gaussian [2] and Lorentzian, or multi-modal), and statistical analysis methods to test parameter collinearity and normality in the residual distribution. Example applications are reported on: a set of data acquired on Zr-1%Nb samples with a Netzsch LFA 467 HT and a Linseis LFA 1000 instruments; UO2 data acquired with a custom-made LFA setup; and W data acquired on a Linseis LFA 1000 instrument. Potential caveats associated with instrumental error are discussed. Finally, the application of PULsE to a participating medium [3] (a type of transparent medium

where thermal radiation is emitted, absorbed and scattered simultaneously) is discussed in relation to the 'established' models and to nuclear fuel at very high temperatures.

References
1. Lunev, A., Zborovskii, V., Aliev, T., Heymer, R., & Vilkhivskaya, O. (2020). PULsE: An open-source software for laser flash analysis. Software Impacts, 6, 100044.
2. Lunev, A., & Heymer, R. (2020). Decreasing the uncertainty of classical laser flash analysis using numerical algorithms robust to noise and systematic errors. Review of Scientific Instruments, 91(6), 064902.
3. Lunev, A., Zborovskii, V., & Aliev, T. (2021). Complexity matters: Highly-accurate numerical models of coupled radiative–conductive heat transfer in a laser flash experiment. International Journal of Thermal Sciences, 160, 106695.
4. Lauerer, A., & Lunev, A. Experimental evidence of gas-mediated heat transfer in porous solids measured by the flash method. International Journal of Thermal Sciences, 184(2), 107948.

308

## Analysis of Strain Localization in Irradiated Austenitic Stainless Steels using FFT-based Crystal Plasticity Simulations

Amirhossein Lame Jouybari[1,2], Samir El Shawish[1], Leon Cizelj[1,2]

[1] Institut "Jožef Stefan", Slovenia

[2] University of Ljubljana, Slovenia

Amirhossein.Lame@ijs.si

The harsh environment of Light Water Reactors (LWR) deteriorates the mechanical properties of internal structures within the reactor. Austenitic stainless steels are among the best structural materials to operate in the LWR environment due to their excellent mechanical properties, and high resistance to stress-corrosion cracking and irradiation damage. However, extended use of these steels may give rise to a unique deformation mode characterized by localized strain in terms of slip and kink bands (also known as clear channels). Such strain localization may lead to a premature strength reduction of the internal components.

The Fast Fourier Transform (FFT) method for the homogenization of composites under periodic boundary conditions is based on the discretized Lippmann-Schwinger equation and Fourier series, and was first developed in 1998. Since then, this method has been extended to polarization-based methods, Krylov approaches, Fourier–Galerkin, and displacement-based methods and pushed beyond a variety of studies containing multi-scale modeling, multi-physics problem, crystal plasticity, etc.

This paper proposes a novel FFT algorithm in the framework of finite deformation crystal plasticity in terms of modified Green operators and Anderson acceleration to acquire a converged displacement field of irradiated polycrystals under tensile loading. Furthermore, the algorithm employs special indicators to identify regions of strain localization within the polycrystalline aggregate, and distinguishes kink bands from slip bands by carefully analysing crystal lattice rotations.

309

# Digital radiography : alternative RT technique for vessel inspection

Severine Annaval

Intercontrole, France

severine.annaval@framatome.com

Digital radiography by a photon counting detector: innovative alternative to RT technique for vessel dissimilar welds inspection.

The RaNADyn project, which was funded by the French Government as part of the recovery plan, aims to develop an innovative digital radiography solution by counting photons, with real-time image, for in service inspection of reactor vessels bimetallic welds (LBM).

In partnership with the Swedish company Direct Conversion, IC has developed a new photon counting detector (PCD) for applications concerning high thicknesses of steel (> 60mm) and therefore requiring the use of High Energy (HE) sources (> 200keV).

The technology has been validated by numerous tests on mock-ups and, in 2020, for the first time on a nuclear site in real conditions. These promising results are the basis of the RaNADYN project, which aims to integrate the HE PCD into a global and industrial system to improve the efficiency of inspection processes thanks to better traceability of controls, a reduction in inspection time, while reducing staff dosimetry too.

The project is based on the following axes:

- Improve the ergonomics of the HE PCD prototype by making it lighter and less bulky to facilitate its implementation into a moving tooling;
- Develop a robotic system, which uses three HE PCDs rotating at constant speed around the pipe during gamma-ray exposure in panoramic configuration;
- Increase data quality by: (i) correcting geometric blur by developing a deconvolution algorithm, and (ii) adding, locally, 3D information by applying a technique close to tomosynthesis. This will allow flaws to be better characterized.
-

We report here the results to date of developments in progress.

310

# Differential thermal analysis of gamma-irradiated nano titanium carbide particles

Elchin Huseynov, Raisa Hakhiyeva

Institute of Radiation Problems of Azerbaijan National Academy of Sciences, Azerbaijan

elchin.h@yahoo.com

In recent years, several modified titanium carbide compounds have become widely used in a wide range of modern technical equipment [1-2]. These sorts of compounds' favourable physical characteristics, such as their resilience to high temperatures, led to an extension of their potential application areas [3-4]. Titanium carbide is a very cost-effective material for equipment and devices that operate at high temperatures. It is quite intriguing to investigate the thermal stability of these types of compounds when used at various temperatures.

Due to the analyzes before and after gamma irradiation, it is known that nanocrystalline TiC particles have stable physical properties up to a temperature of about 1300K by the influence of ionizing radiation. Based on the temperature dependencies of the specific heat capacity, it was found that the nanocrystalline TiC particles were partially oxidized by the influence of gamma radiation and temperature. Therefore, the numerical value of the specific heat capacity of nanocrystalline TiC particles decreases by about three times at the maximum value of temperature.

The temperature dependencies of the specific heat capacity for TiC nanocrystals were investigated for both heating and cooling processes. The positive numerical value of the specific heat capacity in the entire temperature range of the experiments corresponds to endothermic processes. In the cooling process, the numerical value of the specific heat capacity changes little compared to the heating process. In both heating and cooling processes, a decrease in the numerical value of the specific heat capacity is observed under the influence of gamma radiation. This means that relatively high doses of gamma radiation have a direct effect on the heat transfer processes in these materials.

Xinlong Yu et al. (2022) "Microstructures and properties of nickel-titanium carbide composites fabricated by laser cladding" Ceramics International 48, 21814-21821
Xiaoliang Sun et al. (2022) "In-situ formation of titanium carbide in carbon-rich silicon oxycarbide ceramic for enhanced thermal stability" Journal of the European Ceramic Society 42, 6935-6941
Chong Fu et al. (2017) "Fabrication of graphene/titanium carbide nanorod arrays for chemical sensor application" Materials Science and Engineering: C 72, 425-432
Ranjan Dash et al. (2006) "Titanium carbide derived nanoporous carbon for energy-related applications" Carbon 44, 2489-2497

311

# Effect of surface machining on the environmentally-assisted cracking of Alloy 182 and 316L stainless steel in light water reactor environments – results of the collaborative project MEACTOS

Bojan Zajec[1], Tim Austin[2], Rik-Wouter Bosch[3], Mary Grace Burke[4], Michael Grimm[5], Matthias Herbst[5], Anna Hojna[6], Tadeja Kosec[1], Andraž Legat[1], Agostino Maurotto[7], Jill Meadows[8], Radek Novotny[2], Valentin Olaru[9], Thomas Pasutto[10], Francisco Javier Perosanz Lopez[11], Zaiqing Que[12], Stefan Ritter[13], Veronica Román Flórez[14], Alberto Sáez Maderuelo[11], Fabio Scenini[4], Aki Toivonen[12], Aleksandra Treichel[13], Marc Vankeerberghen[3], Liberato Volpe[4], Krystian Wika[7], Mariia Zimina[6]

[1] Slovenian National Building and Civil Engineering Institute [ZAG], Slovenia

[2] JRC, Petten, The Netherlands

[3] SCK CEN, Mol, Belgium

[4] University of Manchester, MPC, Manchester, UK

[5] Framatome GmbH, Erlangen, Germany

[6] CVR, Řež, Czech Republic

[7] NAMRC, Sheffield, United Kingdom

[8] Jacobs, Warrington, United Kingdom

[9] RATEN ICN, Pitesti, Romania

[10] EdF, Moret sur Loing, France

[11] CIEMAT, Madrid, Spain

[12] VTT, Espoo, Finland

[13] PSI, Viligen, Switzerland

[14] ENSA, Maliano, Spain

bojan.zajec@zag.si

The main objective of the EU-funded project MEACTOS (Mitigating Environmentally-Assisted Cracking Through Optimisation of Surface Condition) was to gain knowledge on the ability of different surface machining procedures to mitigate environmentally-assisted cracking (EAC) in some typical light water reactor structural materials and environments.

The surface of austenitic stainless steel (SS) type 316L (cold-worked) and Ni-based weld metal Alloy 182 specimens have been machined in different ways (ground: RS, face milling: STI, face milling in supercritical CO2: SAM1, SAM1 + minimum quantity lubrication: SAM2, shot peening: SP). The EAC initiation susceptibility of these specimens was first screened by accelerated constant extension rate tensile (CERT) tests under simulated boiling (BWR) and pressurized water reactor (PWR) conditions and then verified by constant load experiments.

Scatter in the results of the accelerated EAC initiation testing limited the trends that could reliably be observed, whereby only minor or even no clear improvements of surface grinding (RS) or advanced machining (SAM) compared to the standard industrial face milling were revealed. While the results from the constant load tests confirmed the stress thresholds for EAC initiation in most cases, a fully conclusive picture of the EAC initiation behaviour for all materials and conditions has yet to emerge. In the current presentation, the most important results and conclusions from this five-year collaborative project are summarized.

312

## Thermal creep of MOX fuel: a review of correlations

Rolando Calabrese

ENEA, Italy

rolando.calabrese@enea.it

MOX is the reference fuel for most fast neutron reactor demonstrators and prototypes in Europe. Use of MOX fuel makes licensing procedures of innovative systems easier as an extensive irradiation experience on this type of fuel has been gained in Phénix and Superphénix reactors. Fuel creep rate is an important quantity when estimating Pellet-Cladding Mechanical Interaction (PCMI). Experimental measurements confirm that, beside temperature and stress, plutonium concentration, porosity, grain size and stoichiometry affect this quantity. This paper presents a review of experimental findings and correlations published in the open literature. Correlations are compared and their application in fuel performance codes discussed.

313

## Corrosion behavior of materials in SCW environment

Jana Rejkova[1], Daniela Marusakova[1,2], Marie Kudrnova[1]

[1] UCT Prague, Czech Republic

[2] Centrum výzkumu Řež Ltd., Czech Republic

petruj@vscht.cz

The use of supercritical water in energy technologies is limited by the lifetime of the structural materials. Therefore, it is crucial to obtain specific data on the behavior of suitable materials in this corrosively aggressive environment. In this work, the exposure of several selected materials (T24, 316L, and 800H) under supercritical conditions (450°C, 22.1MPa) in an experimental loop for 500h is described. After the experiment, the surface state and composition were evaluated by XPS (x-ray photoelectron spectroscopy) and SEM (scanning electron microscopy)/EDS (energy dispersive spectroscopy) methods.

# *Thermal Hydraulics and CFD*

401

# TRACE Analysis of Total Loss of Feedwater in PWR

Andrej Prošek, Boštjan Končar

Institut "Jožef Stefan", Slovenia

andrej.prosek@ijs.si

After Fukushima Dai-ichi accident, WENRA (Western European Association of Nuclear Regulators) and the International Atomic Energy Agency (WENRA) require the consideration of design extension conditions (DEC). The purpose of this paper is to study total loss of feedwater (TLOFW) initiating event, which is recognized by IAEA and WENRA documents as possible DEC.

For the analysis, the U.S. Nuclear Regulatory Commission TRAC/RELAP Advanced Computational Engine (TRACE) computer code is used. The TRACE input deck has been developed by applying the conversion of the verified and validated RELAP5 standard input deck for a two loop pressurized water reactor (PWR). For automatic conversion, the Symbolic Nuclear Analysis Package (SNAP) has been used, nevertheless required several manual corrections were still required. The initiating event for the loss of all feedwater is multiple failure, in which all main and auxiliary feedwater pumps are lost. At the DEC initiation, the loss of offsite power (LOOP) was assumed, which means that only safety systems remained available, what is in an agreement with IAEA specific safety guide for deterministic safety analysis. LOOP assumption resulted in the early reactor trip on low reactor coolant system flow, which has very positive impact for TLOFW. Namely, without the LOOP assumption, all normal operation systems would be available and the reactor trip would occur on low-low steam generator narrow level, few tens of seconds after the accident initiation. Therefore, a scenario with assuming reactor coolant pump running until reactor trip on low-low steam generator narrow level and a parametric study varying reactor trip time delay has been performed too. The reactor trip time delay variation was done in 10 s increments up to one minute. In this way the impact the reactor trip time delay on the available time before the reactor vessel level would drop below the level criterion has been assessed. The level criterion was to maintain the reactor vessel level with at least 60.96 cm (two feet margin) above the top of the core.

The results showed that delayed reactor trip reduced the time available before the reactor vessel margin is lost. As expected, the TLOFW scenarios also showed the need for a DEC safety feature. Finally, it has been demonstrated that in the initial 30 minutes there is no need to start alternate auxiliary feedwater pump as a DEC safety feature and that starting the alternate auxiliary pump at 30 minutes successfully prevents the core overheating.

402

## Analysis of an innovative passive heat removal system for station blackout scenario

Sinem Cevikalp Usta, Michael Buck, Joerg Starflinger

University of Stuttgart, Germany

sinem.cevikalp-usta@ike.uni-stuttgart.de

Passive safety systems have the potential to either replace or complement active systems as part of an overall strategy to prevent and/or mitigate nuclear accidents. Additionally, these systems serve as the foundation for many small modular reactors (SMRs), which currently hold a significant position in the nuclear market. The strength of these systems is derived from simple physical principles, such as natural circulation. Consequently, unlike active systems, which rely on external power sources and human intervention to function correctly, passive safety systems are considered more reliable and resilient.

This paper investigates the station blackout (SBO) scenario for an innovative passive safety system design concept of SCOR (Simple COmpact Reactor) design, using a generic dataset. The calculations were performed using the extended version of the thermal hydraulic system code ATHLET for bayonet heat exchangers (BHXs). In this reactor design, residual heat on the primary circuit is removed by RRP (Residual Heat Removal on Primary circuit) loops that contain BHXs. Simulations were performed for two scenarios: one with four active RRP loops cooled by heat exchangers immersed in the pool, and the other without any safety system serving as a reference case. The ATHLET calculation results are investigated for steady-state condition and transient behavior during the SBO scenario. Qualitative comparisons were made with the results of CATHARE calculations from the literature. On the other hand, due to the limited heat capacity of water in the pool, some design proposals have been made to increase its capacity, as there can still be a risk of core melting after a certain period with water-cooled types of RRPs.

403

## Use of CFD for fuel handling and storage safety

Nicolas Goreaud, Emilie may, Adrien Collin de l'Hortet

Framatome, France

nicolas.goreaud@framatome.com

The handling and storage of spent fuel is an activity very demanding in terms of safety, response to public expectations and plant operation efficiency.

Strict safety criteria apply, which can have strong consequences in terms of handling system design and workforce organization.

In order to optimize the design and provide an accurate assessment of the actual margins, Framatome has developed a 3D approach based on CFD and multi scale modelling in order to model accurately and in full details :

- The thermics of the fuel inside the spent fuel cask when it reaches the plant (various situations studied)
- The thermics of the fuel when it is handled in air during its travel between the cask and the pool
- The thermics of the spent fuel pool.

Studied configurations cover normal operation as well as upset conditions (fuel assembly blocked during handling for a variety of reasons) and accidental scenarios (occurrence of a fire, earthquake).
Care has been taken in the definition of the 3D model to enable fast and smooth running, through the use of a multi scale approach :

- Detailed model of portions of fuel assembly enabled to assess mixing, pressure drop and heat evacuation by the fuel assembly
- A dedicated model of fuel assembly is developed
- This equivalent model is simple enough to be added to detailed models of spent fuel cask or full storage pool.

The analysis is performed in 3D and transient (when required), taking into account radiation of heat, ambiant fluid (air or water) convection and conduction inside the structures.

Using such tools, it is possible to assess temperature and air or water velocities inside the room or pool for any situation. It is then possible to perform design optimization in order to ensure an appropriate behavior of the system at the optimal cost.

The use of state of the art, validated, CFD software enables to account for the real geometry and location of all the heat removal systems and take benefit from it. Robustness analysis enable to use the detailed results of 3D analysis as a basis for licensing.

---

404

## Application of CFD and system codes to simulation of HERO-2 experiment

Adam Kecek, Ladislav Vyskocil

UJV Rez, a. s., Czech Republic

adam.kecek@ujv.cz

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 experiments are studied both with system and CFD codes. One of them is the HERO-2 experiment, which simulates heat removal from bayonet tubes through piping and condenser submerged in water pool. In this paper, both open extended test and several closed loop test are used for evaluation of the code capabilities. The CFD simulations are conducted with Neptune_CFD and Ansys CFX codes, where different setup and modelling approaches will be discussed. System code simulations are done with ATHLET 3.3, which

is a part of the AC2 code package. The summary discusses existing challenges and drawbacks of the applied approaches and suggests areas of future code development.

405

## Lagrangian simulation of flow boiling experiments in horizontal annulus

Boštjan Zajec, Leon Cizelj, Boštjan Končar

Institut "Jožef Stefan", Slovenia

bostjan.zajec@ijs.si

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, so 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.

407

## Sensitivity studies on Reverse Natural Circulation in the PKL facility by using RELAP5 mod3.2mz

Qingling Cai[1,2], Vincenzo Zingales[1], Francesco Auria[1], Dominique Bestion[3], Klaus Umminger[4]

[1] University of Pisa, Italy

[2] Xi'an Jiaotong University, China

[3] Consultant, France

[4] Consultant, Germany

qingling_cai@outlook.com

Prediction of pressure drops by wall friction and at geometric discontinuities is important in nuclear thermal hydraulics simulation. Pressure-drop models were established and upgraded for decades; however, errors in predictions of experimental data are up to 40%, namely in low flow and two-phase conditions and particularly

at geometric discontinuities. In order to clarify the impact in situations of interest for nuclear reactors, noticeably accident scenarios investigated in scaled experimental facilities, a virtual benchmark on Reverse Natural Circulation (RNC) in the PKL test facility is being carried out (Ref. [1]), (Ref. [3]).

Since asymmetries between loops and bifurcations occurred in such virtual RNC, in this article sensitivity studies on the impact of the elimination of the loop seal, the impact of the adopted time step and the impact of the nodalization details have been performed by using RELAP5mod3.2mz.

408

## Experimental Investigation of Taylor Bubble Decay Rate in Counter-current flow

Žiga Perne, Blaž Mikuž

Institut "Jožef Stefan", Slovenia

ziga.perne@gmail.com

During the loss of coolant accident (LOCA) in a pressure water reactor (PWR) the depressurization in the primary loop can cause boiling of cooling water and formation of large vapour slugs. In such event, Taylor bubbles may occur in the emergency core cooling system (ECCS) of a pressure water reactor (PWR) or in the U-pipes of a steam generator. In general, Taylor bubbles are unwanted in the primary loop of a PWR, thus, it is important to understand their behaviour in different fluids and flow conditions. We have investigated Taylor bubble decay rate in a vertical pipe with downward liquid flow in order to study the effects of the absolute pressure, temperature, bubble length and pipe diameter on the rate of bubble size reduction. For water-air mixture, it has been observed that a Taylor bubble disintegrates (i.e. decays in size) over time when exposed to a turbulent flow regime. The experiment was carried out in circular glass pipes with inner diameters of 12.4 and 26.0 mm, and 1.6 m in length. Temperature and absolute pressure were recorded at the top and bottom of the test section, and the pressure drop across the pipe was recorded using a differential pressure transmitter (DPT). The bubble size was calculated from the camera recordings along the entire duration of the experiment.

It was observed that longer Taylor bubbles reduce in size faster than smaller bubbles. There are two trends present in regards to the bubble decay rate: longer bubbles seem to decay linearly, while shorter bubbles decay exponentially. This is believed to be a consequence of two mechanisms that determine the decay rate: (a) the physical brake-up of the bubble at the bubble tail into smaller bubbles that get washed away with the flow, and (b) the dissolution of the gas phase into non-saturated liquid phase. The former mechanism is most prevalent in longer bubbles, but it slows down with the reduction of bubble length. Contrarily, the bubble dissolution is always present in non-saturated liquid, however, it is more significant for short bubbles. Subsequently, two sets of measurements were performed: one set for long and one for short bubbles in order to separate the two mechanisms. It was also observed that temperature and absolute pressure affect the decay rate: higher pressure and lower temperature correlate with a faster bubble decay rate.

409

## Sub-channel analysis of the MYRRHA reactor with OpenFOAM: Numerical assessment of turbulence models

Martina Di Gennaro

Politecnico di Milano, Department of Energy, Nuclear Engineering Division, Italy

martina.digennaro@polimi.it

The heavy liquid metal Lead-Bismuth Eutectic (LBE) is the primary coolant of the MYRRHA reactor, in view of its high heat removal capability, low melting point, non-violent reactivity to water and low neutron absorption. One of the phenomenon to be considered for the operation of the reactor with LBE-coolant is the erosion/corrosion of the fuel pin cladding. For this reason, it is key to provide accurate estimations of the cladding outer temperature and LBE velocity during operation. In this work, I investigate the thermo-hydraulic behaviour of LBE in the MYRRHA IPS subchannel (i.e., in-pile test section, internal channel, core design version 1.8), with the perspective of constructing boundary conditions for fuel pin thermo-mechanical analyses. One of the main challenges in the numerical simulation of liquid metals flow with low Prandtl number (around 0.025 for LBE) is to establish a reliable turbulent heat transfer modelling, not being possible to rely on the Reynolds analogy based on the direct proportionality between momentum and thermal boundary layers. A comparative study of different turbulence models is presented to select the most suitable one for studying the low-Prandtl turbulent flow of LBE in the MYRRHA hexagonal rod bundle characterized by a pitch-to-diameter ratio of 1.28. To this end, the Reynolds-average simulation (RAS) models available in the open-source fluid dynamics software OpenFOAM have been applied and assessed against a large eddy simulation (LES) model. The comparison focuses on key thermal-hydraulic parameters, such as the sub-channel temperature field, velocity field, pressure drops, and Nusselt number during reactor normal operation.

410

## Extrapolating Phenomena Observed with Carbon Dioxide to Supercritical Water Reactor Conditions

Walter Ambrosini, Sara Kassem, Andrea Pucciarelli

University of Pisa, Italy

walter.ambrosini@unipi.it

Supercritical Water Reactors (SCWRs) are the only Generation IV concept based on the long-term operating experience achieved with light water reactors. As such, the technological leap for achieving an advanced reactor concept may appear more feasible than for other technologies, in front of advantages as the higher energy conversion efficiency and a more compact design with respect to pressurized water reactors. Studies are being performed since decades in this field, in the aim to achieve a better knowledge of the very peculiar heat transfer and dynamic phenomena occurring when fluids at pressures beyond the critical threshold approach the pseudo-critical conditions, being the locus of rather sharp local maxima of the specific heat,

marking also a dramatic change in other thermodynamic properties. The transition from liquid-like to gas-like conditions, i.e., from a higher to a lower density fluid remaining substantially single-phase, add considerable complexity to a behaviour that was initially believed simpler than at subcritical pressure, owing to the lack of phase change with the related phenomena of departure from nucleate boiling. In fact, in change or as a sort of reminder of the boiling crisis occurring at subcritical pressures, fluids beyond the critical pressure exhibit a departure from the normal heat transfer expected for a single-phase fluid that, especially in mixed convection conditions occurring in heated ducts in upward flow, appears in the form of heat transfer deterioration. Phenomenological analogies between flow boiling and forced convection conditions at supercritical pressures have been established for both heat transfer and flow dynamics, benefitting of modelling techniques applicable to both cases.

In past research activities, the authors of this paper have investigated both the flow dynamic and the heat transfer aspects of fluids at supercritical pressures, developing fluid-to-fluid similarity theories applicable to flow stability and to the various heat transfer regimes that are encountered in experimental tests. The development of a CFD model based on an algebraic expression of the turbulent heat flux available in literature (namely the Algebraic Heat Flux Model) has been an important step in providing a means for predicting observed heat transfer phenomena with unprecedented accuracy with respect to some experimental data. The model was successfully applied to supercritical carbon dioxide data collected at the University of Ottawa (Kline, 2017), which revealed particularly useful for describing a full range of heat transfer phenomena occurring when a fluid at supercritical pressure undergoes heating starting the liquid-like region and reaching the gas-like one, as it is expected for water in SCWR fuel channels. In particular, the data by the University of Ottawa involve normal heat transfer, the start of heat transfer deterioration, a full deterioration to a higher wall temperature manifold, very similar to a sort of boiling crisis, and a final recovery of turbulence occurring when the density in bulk becomes so similar to the one at the wall that buoyancy effects causing flow laminarisation are no more effective. Such a range of behaviours is not easily found in water experiments, owing to temperature and safety limitations, often suggesting to avoid overcoming the threshold of pseudocritical temperature in the bulk fluid during experimental activities.

Thanks to the predictive accuracy achieved in the developed CFD model and to the knowledge about heat transfer phenomena acquired by its application to actual experimental data with different fluids, it is now possible to envisage the behaviour expected to occur in a realistic SCWR subchannel configuration, exhibiting features in close similarity to those observed in CO2 experiments, duly transposed to the operating domain of a supercritical water nuclear reactor. The understanding gained by the experiments and transposed to water by the predictive capabilities of the CFD model provide a synergic and revealing description of expected phenomena, in a convincing picture that is proposed in the present paper for reflection about the heat transfer behaviour that is envisaged in nuclear reactors. This behaviour should be confirmed by experiments that hardly exist at the moment, being anyway necessary for a safe design of SCWRs. The paper, in addition to the description of these expected phenomena, provides also hints for conducting such experiments with water and surrogate fluids, opening a window of reliable predictions of experiments to be performed in future research activities.

Relevant References

Buzzi, F., Pucciarelli, A., Ambrosini, W., 2019. On the mechanism of final heat transfer restoration at the transition to gas-like fluid at supercritical pressure: A description by CFD analyses. Nucl. Eng. Des. 355, 110345.

Kassem, S., Pucciarelli, A., Ambrosini, W., 2021. 2021, Insight into a fluid-to-fluid similarity theory for heat transfer at supercritical pressure: results and perspectives. Int. J. Heat Mass Transf. 168, 120813.

Kline, N., 2017. An experimental study on heat transfer deterioration at supercritical pressures. MSc thesis. Ottawa-Carleton Institute for Mechanical and Aerospace Engineering, University of Ottawa, Ottawa Canada.
Pucciarelli, A., Ambrosini, W., 2017, Improvements in the prediction of heat transfer to supercritical pressure fluids by the use of algebraic heat flux models, Ann. Nucl. Energy, Volume 99, January 2017, Pages 58-67.

411

## Development of surrogate models for quick estimation on debris bed coolability

Markus Petroff, Michael Buck, Joerg Starflinger

Institute of Nuclear Technology and Energy Systems (IKE), Germany

markus.petroff@ike.uni-stuttgart.de

Severe accidents of light water reactors with loss of coolant can lead to overheating of the fuel rods, loss of core integrity and core displacement into the lower plenum of the rector pressure vessel. Due to the possible presence of residual water a debris bed can be formed by fragmentation. The removal of the decay heat is crucial to prevent damage to the rector pressure vessel and further progression of the severe accident. Even though COCOMO-3D is a state-of-the-art mechanistic simulation tool there is a demand for fast running models with less computational cost when coupled in system codes. Instead of using highly simplified models surrogate models have been successfully applied to model complex physical processes which can occur during severe accidents.

Therefore, multiple surrogate models are developed to quickly estimate the dryout heat flux for varied bed configurations i.e. particle diameter, system pressure and bed porosity. As a first step a surrogate model is developed based on a large database generated from various one-dimensional dryout calculations by varying porosity, particle diameter and system pressure using artificial neural networks. In the next iteration a SM is developed based on a database generated by various simple two-dimensional COCOMO simulations. The artificial neural networks are then applied to estimate the dryout heat flux for various bed configurations and validated against the one-dimensional dryout model, COCOMO simulations and experimental data.

412

## Sensitivity analyses by a CFD model on water-wall behaviour in a LW SMR during SBO conditions

Alessandro De Angelis[1], Clément Liegeard[2], Walter Ambrosini[1]

[1] University of Pisa, Italy

[2] CEA Cadarache, France

walter.ambrosini@unipi.it

Small and Modular Reactors (SMRs) are presently among the most relevant subjects of research in the field of innovative nuclear reactors, for the perspective to complement large scale reactors in key areas for the decarbonisation of the energy sector, as the coupling with renewable energy sources in hybrid energy systems and the capability to replace old fossil fuelled plants with an environmentally safe and benign energy source. The characteristics of SMRs as safe and flexible plants to support the energy transition are therefore presently investigated with renewed impulse, in a lively innovation environment in which IAEA has recently recognised several tens of different concepts being proposed worldwide.

The European scenario, in particular, is showing a new interest for the characteristics of SMRs and important political decisions are being taken at the national and the European Union levels. In this frame, a remarkable step was taken on April 4 2023 with an ambitious Declaration on 'EU Small Modular Reactors (SMRs) 2030: Research & Innovation, Education & Training' signed by Commissioner Mariya Gabriel and EU nuclear stakeholders, among which nucleareurope, the Sustainable Nuclear Energy Technology Platform (SNETP), the European Nuclear Society (ENS) and the European Nuclear Education Network (ENEN). The declaration suggests that the European Union is committed "to lead research, innovation, education and training for the safety of European SMRs in support of the EU pre-partnership on SMRs".

As a forerunner in this innovative scenario, the EU ELSMOR Project (Towards European Licencing of Small Modular Re-actors) seeks to design methods and tools for stakeholders to assess and verify Light Water Small Modular Reactor (LW SMR) safety when installed across Europe. In this frame, collection and dissemination of data on LW SMRs is among the actions performed in the project. In particular, in Work Package No. 4 of the project analysis methods and tools for the safety demonstration of improved or innovative containment safety function features of integral SMRs have been developed and assessed. Furthermore, in Work Package no. 5 safety methodologies developed in the project are applied to the E-SMR with reference to specific accident scenarios, as the loss of the normal cooling system.

In the frame of a doctoral research on design and safety aspects of SMRs for the decarbonisation of the energy sector in Europe, a contribution has been given to the ELSMOR project by addressing by CFD techniques the behaviour of containment pools adopted as water-wall systems for rejecting the decay heat in case of postulated accidents. In particular, in a recently published archival paper, water-wall was considered addressing available experimental data and variously scaled systems considered by CFD analyses, in order to extrapolate the behaviour observed in small scale facilities to the full-scale reactor size. Since studying the full size of the reactor containment easily results in unaffordable CFD problems, even for RANS techniques, the methodology adopted in the work consisted in using both 2D and 3D geometries, addressing different scales and carefully comparing the obtained results to judge about their scalability. In addition, a simplified model of the pool obtained by the CATHARE 3 system code was used to compare the overall behaviour obtained by a lumped parameter representation of the pool with detailed CFD calculations.

In the present paper, the previously obtained results are extended by a further sensitivity analysis on the representation of turbulence in the water pool, also addressing a Station Black-Out (SBO) scenario with different turbulence models and numerical advancement schemes. CATHARE 3 is again used to assess the observed behaviour with a simplified reliable model. The additional knowledge of the addressed phenomena thus obtained by these new analyses allows drawing more sound conclusions on the behaviour of containment systems equipped with water-wall systems to be adopted in European SMRs presently under development.

Relevant References

Advances in Small Modular Reactor Technology Developments A Supplement to: IAEA Advanced Reactors Information System (ARIS) 2020 Edition, IAEA 2020, https://aris.iaea.org/Publications/SMR_Book_2020.pdf

Alessandro De Angelis, Nils Reinke, Walter Ambrosini, Assessing water-wall behaviour for a light-water Small Modular Reactor with the aid of CFD analyses, Annals of Nuclear Energy 184 (2023) 109672

Commission Declaration on 'EU Small Modular Reactors (SMRs) 2030: Research & Innovation, Education & Training', https://research-and-innovation.ec.europa.eu/news/all-research-and-innovation-news/commission-declaration-eu-small-modular-reactors-smrs-2030-research-innovation-education-training-2023-04-04_en

EU ELSMOR Project Website, Towards European Licencing of Small Modular Reactors, https://cordis.europa.eu/project/id/847553/it

413

## Development of a 3D Extended Finite Element Model for Efficient Fracture Mechanics Analyses of Hollow Cylinders with Cracks

Nejc Kromar, Oriol Costa Garrido, Leon Cizelj

Institut "Jožef Stefan", Slovenia

nejc.kromar@ijs.si

Fracture mechanics is an ever-important field of study to improve the structural integrity of engineering components with cracks. One of the ways to accurately assess fatigue damage or fracture occurrence is by three-dimensional (3D) deterministic fracture mechanics analysis. The 3D finite element models (FEM) that are used require detailed and dense meshing around the locations of cracks, making long-transient simulations, or simulations of large models with relatively small cracks, very time consuming. Furthermore, engineering components are often complex in shape and make detailed crack meshing difficult. The extended finite element method (XFEM) has been developed to address this issue. This method builds on the conventional FEM by implementing an enrichment that allows the use of discontinuity functions. Because of that, the XFEM can deal with weak and strong computational discontinuities (e.g., material interfaces and cracks, respectively).

This paper presents the feasibility of XFEM for the assessment of internal semi-elliptical cracks in hollow cylinders, such as pipes or pressure vessels, under arbitrary pressure loads. The goal of the paper is to show that to perform fracture mechanics analyses of 3D structural models, the XFEM approach offers accurate results with decreased computational costs and work complexity, since it requires no or minimal re-meshing. To that end, two 3D fracture mechanics models are developed in Abaqus: one that uses XFEM without a detailed mesh and another that uses conventional FEM with a detailed mesh around the postulated crack. The results for stress intensity factors (SIFs) obtained with both models are compared and verified against the results with semi-analytical SIF formulations. For the later, the 3D through-wall stress distribution values are used as inputs. The outcomes of the paper include the reduction in computational time and costs achieved by using XFEM approach, and verification with conventional FEM and established semi-analytical calculations.

414

## Bubble Breakup Sensitivity on Local Surface Tension Modification in LES of Turbulent Slug Flow

Jan Kren[1,2], Iztok Tiselj[1,2], Blaž Mikuž[1]

[1] Institut "Jožef Stefan", Slovenia

[2] University of Ljubljana, Faculty of Mathematics and Physics, Slovenia

jan.kren@ijs.si

Understanding the complex behavior of two-phase flow during accidental conditions in nuclear power plants requires evaluating flow properties under varying conditions. This study aims to investigate the influence of enhanced surface tension models based on the curvature of the bubble interface on the behavior of Taylor bubble. The primary focus is the development and implementation of such models and the detailed analysis of the local surface tension modifications on the bubble breakup dynamics.

Two-dimensional simulations are conducted with the modified interFoam Volume of Fluid (VoF) solver in OpenFOAM. The study provides valuable insights for the development and testing of subgrid-scale models to manipulate bubble coalescence and breakup behavior. Findings reveal that Taylor bubbles in water-air mixture are significantly influenced by surface tension, with enhanced surface tension near the interface leading to decreased bubble breakup and vice-versa. The shape of Taylor bubbles is also notably affected by surface tension, particularly at the bubble nose and tail region. Validation is performed using 3D LES simulations and experimental results in pipes with different diameters and Reynolds numbers.

415

## A mechanistic bubble force model for the development of boiling parameters in high heat flux regimes

Aljoša Gajšek[1,2], Matej Tekavčič[1], Boštjan Končar[1]

[1] Institut "Jožef Stefan", Slovenia;

[2] University of Ljubljana, Faculty of Mathematics and Physics, Slovenia

aljosa.gajsek@ijs.si

Nucleate boiling stands as a highly efficient heat transfer mechanism, capable to remove substantial quantities of energy from a heated surface while only prompting a minimal rise in surface temperature. However, a too high heat flux at the surface can lead to the occurrence of boiling crisis. This phenomenon transpires when the heated surface becomes crowded with emerging bubbles that coalesce into an insulating vapour layer. Subsequently, this is followed by a dramatic increase in the wall temperature, resulting in damage of the heated structures.

An insufficient understanding and prediction of the boiling process and the Critical Heat Flux (CHF) often results in costly and overly conservative solutions. Such designs are purposefully circumventing the boiling process altogether, operating in a single-phase convection mode only, instead of embracing the potential effectiveness of boiling heat transfer. This is also the case in the design of fusion divertors.

The heat flux received by the heated wall serves as an energy boundary condition within the Navier-Stokes equation. A mechanistic boiling model partitiones this heat flux into evaporation, single-phase convection, and quenching. The model then calculates the distribution of heat between liquid heating and vapor formation, a computation dependant on three key boiling parameters: nucleation site density, bubble detachment frequency, and bubble detachment diameter. Understanding and quantifying these parameters is therefore essential for accurate modelling of boiling flow.

In this work we aim to quantify the effects and significance of each force acting on a nucleating bubble under varied operating conditions. A particular emphasis will be placed on the calculation of boiling parameters of bubble detachment frequency and bubble detachment diameter from those forces. These parameters will subsequently be compared to empirical models. Ultimately, our research seeks to mitigate the existing disparity in the modelling of boiling flows in extreme operating conditions.

416

## The 3D DNS simulation of a Taylor bubble in counter-current flow with a turbulent wake using the Front-Tracking method in TrioCFD

Raksmy Nop[1], Alan Burlot[1], Guillaume Bois[1], Blaž Mikuž[2], Iztok Tiselj[2]

[1] Université Paris-Saclay, CEA, Service de Thermo-hydraulique et de Mécanique des Fluides, France

[2] University of Ljubljana, Faculty of Mathematics and Physics, Slovenia

raksmy.nop@cea.fr

The understanding of two-phase flows is of key importance for the nuclear industry as they may occur in normal operation, like in the steam generators, or in accidental scenarios, like in the core during a loss of coolant accident (LOCA). Among the variety of regimes that may occur in a two-phase flow, there is the slug flow composed of different kind of pattern. One of them is the Taylor bubble which is a large bubble filling almost the entire section of the vertical channel in which it tends to rise up. The correct simulation of a Taylor bubble is challenging as it can be decomposed in three zones involving different scales and mechanisms: the upstream zone depending on the inlet conditions with the bubble head which has a bullet shape, the bubble body surrounded by an annular thin liquid film and the bubble tail where the shape and the resulting downstream flow may be laminar or turbulent.

In this communication, we present simulations of Taylor bubble experiments realized by the THELMA laboratory at the Reactor Engineering Division of Jožef Stefan Institute, Slovenia. These experiments were done using air-water mixture in a vertical pipe with a diameter of 12.4 mm and a length of 1500 mm. The Taylor bubble is investigated in a counter-current configuration: the flow is directed downwards with a flowrate allowing the bubble to be stationary for long visualizations. The upstream flow is laminar as the Reynolds number Re=ULd/ν is about 1400, but the bubble wake is expected to be turbulent as the inverse viscosity number Nf=($Eo$3/$Mo$)1/4 is high (about 5x103), with Eo the Eötvös number and Mo the Morton number.

The simulations were realized with TrioCFD, the open-source CFD code developed at CEA. They were performed in Direct Numerical Simulation (DNS) using the Front-Tracking (FT) method for interface tracking. Note that this choice of DNS+FT to simulate a Taylor bubble is particularly scarce in the literature and is a premiere for such conditions (e.g. involving a turbulent wake). A 3rd order Runge-Kutta time-integration scheme is used and the meshes are made of several million tetrahedron elements. In a first step, the simulations are validated using THELMA experimental results such as pressure drop, relative velocity or liquid film thickness. Then, we investigate physical phenomenon like the liquid film velocity profile or the bubble tail turbulent velocity.

417

## Uncertainty and Sensitivity Analysis of Hot Leg Loss of Coolant Accident Simulated by RELAP5

Andrej Prošek

Institut "Jožef Stefan", Slovenia

andrej.prosek@ijs.si

The reactor pressure vessel of water-cooled reactors of non-boiling type is one of the most important and non-replaceable components. Among the risks that threaten the integrity of the reactor pressure vessel is the possible destruction due to pressurised thermal shock (PTS). PTS can occur during several postulated accident scenarios, including loss of coolant accidents. The purpose of this study is to perform the uncertainty and sensitivity analysis for RELAP5 simulation of small break loss of coolant accident, located in the hot leg. Namely, the results of thermal hydraulic calculations are needed for further structural analysis. The reactor selected was two-loop pressurized water reactor (PWR), for which verified and validated input deck was available. The RELAP5 developmental version 33lj from 2022 with built-in code uncertainty parameters has been used in this study. In total 15 uncertain input parameters have been considered. The selection of input uncertain parameters was based on the results obtained in the frame of Advanced PTS Analyses for LTO project (APAL), where LTO abbreviation means long term operation, respectively. For uncertainty and sensitivity anlaysis Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.5 has been used. For uncertainty and sensitivity analysis 130 runs have been performed with RELAP5. It was assumed that loss of offsite power is concurrent with the break occurrence. For the purpose of asymmetry effects, only one (out of two) high pressure and low pressure injection pump were assumed available. The list of input uncertain parameters includes also initial conditions, therefore for each run a steady-state run has been performed. The main figures of merit in uncertainty analysis were reactor pressure, liquid temperature and reactor vessel wall temperature below the cold leg connection. The results showed that figures of merits for reference case are bounded by the results of 130 runs. The sensitivity analysis showed that the most influential parameters are high pressure safety injection system temperature and flow, initial pressurizer pressure, form loss coefficient and thermal non-equilibrium coefficient for Henry-Fauske choke flow model.

418

## Simulation of PASI experiment on passive containment heat removal system with MELCOR and ATHLET

Adam Kecek, Zbynek Parduba

UJV Rez, a. s., Czech Republic

adam.kecek@ujv.cz

Passive safety systems are in focus of several research projects. One of them is the EU PASTELS project, which 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 experiments are studied extensively with different computational tools, including CFD and system codes. Variousnumerical activities have been carried outon experiments in the PASI test facility, which simulates heat removal from containment through passive condenser connected to an external water pool. This paper summarises simulations performed with ATHLET 3.3, which is a part of the AC2 code package and severe accident code MELCOR 2. The capabilities of the codes are discussed, highlighting the existing challenges and drawbacks of the applied approaches. A key part of the paper is a discussion on areas of future code development.

419

## Direct Contact Condensation Induced Water Hammer Simulation Using Computational Fluid Dynamics Code

Ivo Kljenak[1,3], Md Arman Arefin[1,2]

[1] University of Ljubljana, Faculty of Mathematics and Physics, Slovenia

[2] LUT University, LUT School of Energy Systems, Finland

[3] Jozef Stefan Institute, Reactor Engineering Division, Slovenia

ivo.kljenak@ijs.si

Water hammer occurs if a pressure surge, or a high-pressure shockwave, propagates through a piping system when a fluid in motion is forced to change direction or stop abruptly. This may also occur due to direct contact condensation (DCC), resulting from the interaction of steam and subcooled liquid water. DCC-induced water hammer is a very likely phenomenon during accidental events in light water reactors, and could cause significant damage to the piping.

The combination of the very short time scale on which DCC occurs and the random shape of the gas-liquid interface (in the sense that it cannot be predetermined exactly) on which condensation occurs results in an (aleatory) uncertainty of the magnitude of the pressure surge. This means that experimental values of maximum pressure may vary significantly from one test to the next, even if initial and boundary conditions are kept strictly constant. First, this makes it difficult to replicate theoretically experimental values of

maximum pressures. Second, such results of theoretical simulations will always be uncertain (with regard to real phenomena), regardless of the sophistication of the modelling.

In the proposed paper, DCC-induced water hammer was simulated on the local instantaneous scale, with the application of large-eddy simulation. The Computational Fluid Dynamics (CFD) code ANSYS Fluent was used, with a modification introduced via a user-defined function. The model was first validated using an experiment from the literature. Then, the model was applied to DCC-induced water hammer in a T-junction. After the simulation results were compared with results from a similar case in the literature, the model was used to study the influence of liquid water subcooling and mass flow rate on the magnitude of the pressure spike. As might be expected from physical intuition, the simulations revealed that it is possible to avoid water hammer by increasing the liquid water temperature closer to the saturated steam temperature, and that the water hammer dampens with the increase of the liquid water mass flow rate. The developed model enabled also a qualitative analysis of these influences.

420

## RELAP5 Validation Against Experiment Performed in SIRIO Experimental Facility

Rok Krpan, Ivo Kljenak, Andrej Prošek

Institut "Jožef Stefan", Slovenia

rok.krpan@ijs.si

In the framework of the research and development activities dedicated to the development of new and innovative nuclear technologies, different experiments are performed to address new configurations and systems. To investigate one of such topics, the European Commission funded the HORIZON 2020 PIACE project (2019-2022), with the main objective to support the technology transfer from the research to industry in the area of safety of nuclear installations. In particular, the project supported the development of a concept of Passive Isolation Condenser, an innovative decay heat removal system designed to provide a passive safe long-term cooling of nuclear reactor, as well as to passively control the heat removal from the ultimate heat sink by means of a self-controlled injection of non-condensable gases.

The suitability of the concept was tested in SIRIO experimental facility located at SIET in Piacenza (Italy). The SIRIO experimental facility consists of a Steam Generator (SG) and two loops. A loop with a heat exchanger is used to establish steady state conditions, and a loop with a passive Isolation Condenser (IC) is used to replicate the decay heat removal from the reactor core during an accident, using the proposed design. The SG includes electrical heaters and molten salts in a bayonet arrangement, which in turn heats and evaporates liquid water. During the transient, the SG power decreases following a decay heat curve and the steam from the SG secondary side flows into the IC, where it condenses, causing pressure and temperature decrease on the SG secondary side. The introduced innovative concept is that the IC lower head is connected to a tank of non-condensable gas. When the pressure in the IC lower plenum drops below a certain value, the non-condensable gas flows into the condenser tubes and reduces the condensation rate. This lowers the decrease rate of pressure and temperature and reduces the cooling rate of the reactor core.

System codes approved by the regulators are typically used to analyse the transient behaviour after some initial event in existing nuclear power plant systems. Furthermore, these codes are even used in the licensing process of new nuclear builds. To ensure the correct representation of physical phenomena, the equations, the physical models used in the equations, the methods of solutions implemented in such system codes, and also the recommended modelling and nodalization practices must be verified and validated against the relevant experimental data. In the present work, the U.S. Nuclear Regulatory Commission qualified thermal-hydraulic code RELAP5, (MOD 3.3 Patch 5), is being used and validated against the experiment performed in SIRIO experimental facility. To support the experimental campaign, JSI developed an input deck of the SIRIO experimental facility for the RELAP5 system code in order to perform the first set of simulations already before the execution of the experiment. In accordance with the protocol of the experiment, a steady state, where the steam flows from SG through a heat exchanger with constant removed power, was established. The results of the steady state simulation was then used as initial conditions for the simulation of a transient with decreasing SG power and removal of the heat through passive isolation condenser.

The results of these pre-test calculations were presented in earlier works (also at the NENE 2022 conference). After the experiment was performed, the experimental results were compared to the simulation results, and a large disagreement was observed (a pressure spike in the simulation right at the beginning of the transient phase, which did not occur in the experiment). In the present work, firstly, the deficiencies in the modelling approach and in the RELAP5 thermal-hydraulic system code will be identified. Secondly, the input deck will be calibrated to better replicate the behaviour observed in the experiment, thus increasing the prediction accuracy and the understanding of both the best modelling practices and the capability of empirical models, implemented in the system codes, to predict the system behaviour. Lastly, common conclusions (and guidelines) applicable for future applications to natural convection facilities models will be drawn.

421

## Experimental Investigation of Thermal Wave Flow Meter for Molten Salt Applications

Tomas Korinek, Jan Skarohlid

Czech Technical University in Prague, Czech Republic

tomas.korinek@cvut.cz

Flow measurement of molten salts is a challenging task due to high temperatures and high corrosion of salts. A simple and reliable flow meter capable to withstand harsh conditions of molten salt is a key feature for the safe operation of molten salt reactors. The present study experimentally investigates the potential of thermal wave flow meters for flow measurement of molten salt. The flow meter is not limited only to molten salt reactors but can be applied in other applications such as flow measurement in concentrating solar power plants, and thermal storages. Experimental investigations were conducted with water on an experimental flow meter platform with a reference magnetic-inductive flow meter. Several flow and thermal wave parameters were tested in order to study the behavior of thermal waves.

422

# Analysis of Steam Generator Tube Rupture (SGTR) Accident using RELAP5/MOD 3.3 and TRACE 5.0p5 Codes

Vesna Benčik, Davor Grgić, Siniša Šadek

University of Zagreb Faculty of Electrical Engineering and Computing, Croatia

vesna.bencik@fer.hr

NPP Krško (NEK) input deck for best-estimate computer code RELAP5/MOD 3.3 has been developed at Faculty of Electrical Engineering and Computing (FER), Zagreb. Recently, the NPP Krško model for TRACE code based on RELAP5 model has been completed. Currently, for verification purposes for TRACE code, the on-transient qualification is performed by comparing the transient results with RELAP5 code. In this paper the results of Steam Generator Tube Rupture (SGTR) accident for NPP Krško using RELAP5/MOD 3.3 and TRACE 5.0p5 are presented. The SGTR event presents the threat by providing the direct path for primary coolant to the environment via the secondary side relief valves thereby bypassing the containment. Unlike other loss of coolant accidents, an early operator action is necessary to prevent the radiological release to the environment. The primary-to-secondary leakage causes the primary pressure drop which leads to reactor trip either on low pressurizer pressure signal or overtemperature DT signal. Furthermore, the Safety Injection (SI) signal may be generated due to continuous primary pressure decrease. After SI actuation the primary pressure will tend to stabilize at the value where SI flow equals the flow through the ruptured tube. The operator is expected first to determine that the SGTR event has occurred and to isolate the broken SG by closing the main steam isolation valve. The subsequent operator actions consisting of controlled cooldown and depressurization are aimed to stop the primary-to-secondary leakage on one side and on the other side to achieve conditions where Reactor Coolant System (RCS) is cooled via intact SG and the SI is isolated. After achieving these goals the plant cooldown and depressurization to hot shutdown when the Residual Heat Removal (RHR) system can be put in operation is performed.

The postulated accident is a double-ended break of one U-tube located at the tube sheet on the cold leg side. The analyses have shown that this location of the break leads to maximum break flow when reactor core is at subcooled conditions. The operator isolated the broken SG 16 minutes after reactor trip. In the analysis, the loss of offsite power was assumed, i.e., the Reactor Coolant Pump (RCP) trip in both loops as well as the isolation of main feedwater resulted immediately after reactor trip. The steam dump system was assumed unavailable. Thus, the operator used the intact SG relief valve to perform the controlled cooldown. The transient for both RELAP5 and TRACE was simulated for 3000 seconds when the stable conditions were achieved with primary-to-secondary leakage stopped and with RCS cooling performed via intact SG.

423

## Resolution-Adaptive Modelling in Nuclear Safety: Free Surfaces and Bubbles

Matej Tekavčič[1], Richard Meller[2], Benjamin Krull[2], Fabian Schlegel[2]

[1] Institut "Jožef Stefan", Slovenia

[2] Institute of Fluid Dynamics, Helmholtz-Zentrum Dresden–Rossendorf, Germany

matej.tekavcic@ijs.si

Process engineering and energy production systems often feature gas-liquid flows with coexisting two-phase flow regimes and a broad range of interfacial and turbulent scales. Suitable simulation methods can be found for each particular morphology, such as Volume-of-Fluid for larger, resolvable and continuous features of stratified flows, and the two-fluid model for unresolved dispersed bubbles or droplets. A morphology-adaptive multifield two-fluid model (MultiMorph) presented here is developed based on the OpenFOAM Foundation Release, with the aim to handle different coexisting dispersed and continuous flow structures for a wide range of spatial resolutions within a common computational tool.

The present work highlights specifically the following aspects of the model: a resolution-adaptive momentum transfer for under-resolved flow structures on coarse meshes, interface turbulence damping in strong shear flow near a gas-liquid surface with high density ratios between the phases, and morphology transfer models. This enables both transitions, disintegration and accumulation, between dispersed and continuous phase morphologies of the same fluid. Application of the MultiMorph model is presented on the following selected set of safety related test cases: a stratified counter-current flow case with partial flow reversal and liquid waves, and a plunging jet case, with entrainment of gas bubbles. Results are evaluated with measurements from the corresponding experiments.

424

## Loss of Coolant Accident with Total Failure of High Pressure Injection System in Two-loop PWR

Andrej Prošek

Institut "Jožef Stefan", Slovenia

andrej.prosek@ijs.si

After Fukushima Dai-ichi accident WENRA (Western European Association of Nuclear Regulators) and the International Atomic Energy Agency (WENRA) require due consideration of design extension conditions (DEC). The purpose of this paper is to study design extension condition in which total failure of high pressure injection system occurred. Such multiple failure has been recognized by International Atomic Energy Agency (IAEA) and WENRA documents as possible DEC.

For analysis the U.S. Nuclear Regulatory Commission TRAC/RELAP Advanced Computational Engine (TRACE) computer is used. The TRACE input deck has been developed based on the conversion of verified and

validated RELAP5 standard input deck for two loop pressurized water reactor (PWR). For automatic conversion the Symbolic Nuclear Analysis Package (SNAP) has been used, which required also manual corrections. The initiating event total failure of high pressure injection system is multiple failure in which both high pressure safety injection pumps are lost. Other safety systems are assumed available. The TRACE calculations have been performed for a spectrum of break sizes. The results showed that DEC safety features are needed for smaller breaks, because such breaks could not remove all decay heat through the break. When the breaks are larger, the decay heat removal through the break is sufficient as pressure drops sufficiently to allow accumulators and low pressure system injection. Finally, calculations demonstrated that in scenarios assuming DEC features the significant core damage has been prevented.

425

## Measurement of Radiative Transport Properties of Water and Refrigerant R245fa

Mihael Boštjan Končar, Blaž Mikuž, Matej Tekavčič

Institut "Jožef Stefan", Slovenia

mbkoncar@gmail.com

Infrared thermography could significantly contribute to advancing our comprehension of heat transfer phenomena, particularly in the field of nuclear engineering. It enables the non-intrusive experimental study of various heat transfer processes, including flow boiling, where experiments play a crucial role in the investigation of phenomena. However, with thermography, an accurate determination of radiative heat transfer properties is of vital importance to yield reliable results. Presented study is focused on the determination of transmissivity properties of liquids and emissivity of air-liquid interface. Our study entails the analysis of water and the refrigerant R245fa.

The aforementioned properties were studied using a long-wave infrared (LWIR) sensitive high-speed camera, i.e. wavelengths from 7.5 to 12.5 microns. Different liquid temperature conditions were examined. The paper provides a description of the measurement methodology, along with a presentation of the experimental results. The described method was applied to determine the surface emissivity of a polished aluminium surface, which enabled a comparative analysis of the results with values documented in the existing literature, and thus validation of the method. The uncertainty of all the measurements has been evaluated with the error propagation analysis.

426

## Comparison of the main parameters behavior in VVER 1000 during a MSLB accident for different fuel campaigns

Antoaneta Stefanova, Pavlin Groudev

INRNE-BAS, Bulgaria

ant_stefanova@abv.bg

The present paper is focused on investigation of nuclear power plant behaviour parameters in case of „Main steam line break“ (MSLB) accident for different fuel campaigns. The calculations have been performed using RELAP5/mod3.3 computer code.

Most of assumptions applied in preparation of the executed scenario have been developed based on the investigation done in the performed previously OECD VVER-1000 MSLB benchmark problem. The accepted new assumptions have been accepted for simplification of investigation.The main aim of the performed work was to investigate the response of the main plant parameters during MSLB accident. The other objective was to investigate the reactivity response in conditions corresponding of the beginning of fuel cycle of 1st and 8th fuel campaigns of the VVER1000 reactor. Additionally, the important phenomena observed during the MSLB as the reversing of the flow rate in the damaged loop have been analized.

The simulated scenario is “Main steam line break” with ID 580 mm. The break is located outside of the containment, between the SG #4 and “Steam isolating valve” (SIV). The investigated event is characterized with asymmetrical core and vessel cooling and by significant space-time effects. The investigated scenario is important for investigation of core criticality and possible power return.

The performed work is important for code improvements and plant behaviour safety assessment of VVER1000.

427

## EU DEMO upper port transfer cask CFD analysis

Martin Draksler[1], Christian Bachmann[2], Boštjan Končar[1]

[1] Institut "Jožef Stefan", Slovenia

[2] PPPT Power Plant Physics and Technology, EUROfusion Consortium, Germany

martin.draksler@ijs.si

An analysis of the EU DEMO upper port transfer cask with a hot BB segment inside is carried out to determine the temperatures of the BB segment and the cask over the time. Since the cask is filled with air at 95 kPa, the natural circulation of the air is expected to establish, promoting the cooling of the BB segment by transferring the heat to the cask walls. It is assumed that the remote maintenance will take place 1 month after plasma shutdown.

In addition to a transient 3D CFD analysis, a simplified theoretical model, based on the energy conservation law, was developed. Both models were used to predict the time-dependent temperature trends of a BB segment that has been placed into an empty cask. In this study, it is conservatively assumed that the heating power per outboard WCLL BB segment due to decay heat is 10 kW. It is conservatively assumed that the BB temperature is 100°C when it would be picked by the BB transporter.

The CFD analysis described here shows that naturally circulating air in the cask can provide a sufficient cooling mechanism for decay heat removal from undrained BB segment to stabilize its temperature within few days if the external cooling of the cask is sufficiently effective (e.g. either rather strong natural convection is established and/or the air temperature is sufficiently low).

428

## Calculation of HI-STORM FW cooling by natural convection of the air

Davor Grgić, Paulina Dučkić, Siniša Šadek, Petra Strmečki

University of Zagreb Faculty of Electrical Engineering and Computing, Croatia

davor.grgic@fer.hr

Holtec's HI-STORM FW cask is used for dry storage of spent nuclear fuel in NPP Krsko Spent Fuel Dry Storage (SFDS) installation. Cask's body and top lid are forming protective (mechanical and shielding) enclosure for housing of Multi Purpose Canister (MPC) with up to 37 spent fuel assemblies. The decay heat is transferred from spent nuclear fuel to the MPC surface by natural convection of Helium, by conduction through parts of the fuel basket and Aluminum shims, and in the case of elevated temperature by heat radiation. The heat is dissipated from the surface of the canister by natural circulation of the air flowing in space between the MPC and HI-STORM internal surface and, to some extent, by heat radiation. Finally, some heat is transferred from HI-STORM surface, again mostly by natural convection to the air and to smaller extent (in normal operation) by heat radiation. In this paper, a steady state simulation of the air flow and related heat transfer within the cask is performed using ANSYS/FLUENT code. Thermal conditions inside MPC are not explicitly modeled because the focus is on the prediction of air temperatures and surface temperatures of the MPC and HI-STORM for spent nuclear heat loads in the range between 15 and 25 kW (the values found in 16 casks to be loaded in NPP Krsko campaign 1). The air surrounding the cask is a separate volume with appropriate boundary conditions to model natural circulation. The HI-STORM cask has temperature monitoring system used to measure air temperature at the cask inlet and outlet vents, to be a measure of unobstructed air flow. The additional use of the temperatures measured by the system is to be indication of the Helium leakage based on the difference in measured temperatures of MPC top and bottom lid surfaces. The intention of this calculation is to correlate calculated and measured temperatures and to understand reasons for the differences.

# Usage of Monte Carlo Code Serpent2 for Calculation of FHR Fuel Assembly

Davor Grgić, Paulina Dučkić, Radomir Ječmenica, Bojan Petrović

[1] University of Zagreb Faculty of Electrical Engineering and Computing, Croatia

[2] Georgia Institute of Technology, Atlanta, GA, USA

martin.draksler@ijs.si

In this paper our initial results are presented for Fluoride-salt High-temperature Reactor (FHR) reactor physics benchmark calculations, Phase I-C. Phase I-C extends to 3D previous OECD benchmark Phase I-A and I-B, which defined pseudo-2D calculation of a single FHR fuel assembly with TRISO fuel, moderated with graphite and cooled with FLiBe coolant. Pseudo-2D fuel element geometry is extruded in axial direction with addition of axial top and bottom reflectors (FLiBe and graphite). Radially, periodic boundary conditions (BC) were applied, and axially, vacuum BCs were used. The characteristics of the benchmark, complicated 2D geometry of plate type assembly with TRISO fuel, double heterogeneity spectral calculation and use of 'exotic' materials (FLiBe coolant, Eu as burnable poison), mean that the most suitable calculation tool should be Monte Carlo computer code. We used Serpent2 code (versions 2.1.32 and 2.2.1, compiled for Cygwin environment under MS Windows) with three versions of ENDFB library (6.8, 7.0 and 7.1). In addition, CSAS6 module from SCALE 6.2.4 package was used to check obtained keff values. The paper covers the results (keff, fission density spatial distribution, group fluxes, and selected isotopes number densities) of the first 4 benchmark exercises. The first two exercises assume axially symmetric core, with uniform temperature, without depletion. The third exercise analyses control rod insertion and the fourth is the same as the first one, but with depletion up to 70 GWd/tU. Our goal was to check the differences when using two most recent Serpent2 versions (almost the same results and small change in CPU time) and what are differences in results and calculation time when three versions of ENDFB library were used. The decision was to use faster (and less memory demanding) ENDFB 6.8 library for scoping calculation and 7.1 library for production calculation.

# *Fuel Cycle and Radioactive Waste*

501

## Krško NPP Spent Fuel Dry Storage Project – Technology and Implementation Overview

Vid Merljak, Rok Bizjak, Andrej Kavčič

Nuklearna elektrarna Krško d.o.o., Slovenia

vid.merljak@nek.si

As a part of a major nuclear safety update programme and to support operating lifetime extension, Krško NPP decided to implement a Spent Fuel Dry Storage system. In essence, spent fuel assemblies are transferred from actively cooled water pool to a dry storage building, where the decay heat is removed passively by air convection. Fuel assemblies reside in and are additionally protected by airtightly welded casks which become an additional fission product barrier. This dry storage system was put to service in March 2023. To date, a few hundred fuel assemblies have been transferred to storage casks. In this paper, we present technology bases and its implementation at Krško NPP.

502

## How will backend issues affect the global deployment of SMRs?

Charles McCombie[1], Neil Chapman[2], Leon Kegel[3], Jake Kinghorn-Mills[4], Sean Tyson[5]

[1] McCombie Consulting, Switzerland

[2] Chapman Consulting, Ireland

[3] Agency for Radioactive Waste Management, Slovenia

[4] SloveniaMCM Environmental, United Kingdom

[5] Department of Energy, United States of America

charles.mccombie@mccombie.ch

Today, the hopes of the nuclear industry for a revival of interest in expanding capacity are based to a large extent on the expectation that the new small modular reactors (SMRs) that are being developed will be deployed at a large scale globally. SMRs promise numerous benefits, including lower capital costs, enhanced safety features, easier grid compatibility, and other versatile applications extending beyond electricity production. However, there are key questions still to be answered. Will the economy of multiples outweigh the economies of scale which led to NPPs becoming larger? Can the short timescales to implementation be achieved? How will the so-called “unsolved waste disposal problem”, which has been the justification for much opposition to conventional NPPs, be addressed for SMRs. This last question is currently being neglected by many SMR designers and is the focus of this paper, which draws heavily from on an on-going cooperative study sponsored by USDOE and the ERDO Association.

The key back-end issues associated with any nuclear power plant relate to management and disposal of spent fuel and other radioactive wastes arising during operation and decommissioning of the plant. The most challenging tasks are to design, implement and operate a deep geological repository (DGR) or a deep borehole disposal facility (DBDF) since these are the only accepted approaches to achieving a safe solution for disposal of the high-level and long-lived wastes produced by nuclear power plants. The key characteristics of the wastes that determine the repository design are their physical and chemical forms, volumes, dimensions, activity levels and heat production. These have been examined for examples of the main SMR designs being developed.

There are various potential policies and strategies that can be adopted for ensuring that any national nuclear program considering SMR implementation has access to a safe, state of the art disposal facility:

- Develop a national DGR – an expensive facility with a long realisation timescale
- Dispose in a national DBDF – not yet a proven technology
- Dispose in a Multinational Repository (MNR) – no current facility available
- Look for fuel or SMR suppliers offering a "take back" or "take away" service including final disposal – none do so at present
- Keep all options open - safely manage wastes in dedicated facilities and operate a "dual track" (national plus multinational) disposal policy

This paper discusses the merits and the viability of these options.

The discussion leads to conclusions regarding the SMR market. Potential vendors are mostly neglecting backend issues, but potential customers in newcomer countries should highlight these and emphasise their impact on the choice of SMR design and supplier. Countries that hope to become major SMR suppliers could benefit by considering take back and/or supporting MNR proposals elsewhere. It seems clear that wide deployment of SMRs and their regional dispersion will encourage regional or multinational cooperation in all aspects of radioactive waste management, including implementation of MNRs.

503

## Radwaste handover project to ARAO, Fond and Decommissioning of the original steam generators

Stanko Manojlović, Matjaž Gričar

Nuklearna elektrarna Krško d.o.o., Slovenia

stanko.manojlovic@nek.si

For 40 years of operation NEK has put considerable effort in reducing Low and Intermediate Level Radioactive Waste (LILW). In 2000, NEK also replaced the original steam generators (SG) with new Siemens SG's, storing the old ones inside a for a purpose-built Decontamination building (DB).

However, after 40 years of operation the Radwaste Storage Building (RWSB) is near to be filled with operational LILW. As stipulated by the Intergovernmental Agreement (IA), the Republic of Croatia and the Republic of Slovenia must take over each half of the waste within two years after the end of the regular operating period of the power plant (2023-2025), if they do not agree on a joint solution. As known after years of negotiations joint solution has not been agreed (Ref. Minutes of 13. Intergovernmental Commission meeting), therefore LILW needs to be divided and handed over to responsible Agencies in Slovenia and Croatia (ARAO and Fond). None of them has disposal or long-term solutions ready to be used, yet. In addition to that, different technology of disposal/long storage are planned to be used; N2d disposal containers by ARAO and RCC by Fond.

In April 2022 the Intergovernmental Commission agreed that part of the waste, which will not be further processed and sent for reprocessing, is going to be packaged at the NEK location for both, ARAO and Fond. This is to be done by ARAO's and Fond' procedures and on their expenses. For these reasons, ORANO and Siempelkamp NIS studies were prepared, showing whether this is possible to be done and what additional equipment and changes NEK needs to support this process. Some of the important results are shown in this presentation.

To assure long-term operation of the plant, NEK needs to make sure all necessary actions are taken to hand-over the existing LILW and make required free space in LILW storage. Since this is important fourth condition for Long-term operation - LTO (besides Safety Upgrade Project - SUP, Periodic Safety Review 3 - PSR3 and Environmental Consent for LTO), NEK prepared an Action Plan and Risk Assessment to take additional actions. One of them is to release the space where old steam generators are being stored and use it as storage for uncompressible LILW packages, as foreseen by Construction Permit for the Decontamination Building. However this project is very demanding and due to number of reasons still uncertain.

504

## Calculation method for determining neutron-induced nuclide activities in nuclear facilities

Sebastian Schoop, Mehmet Kadiroglu, Kai-Martin Haendel

TÜV NORD EnSys GmbH & Co. KG, Germany

Sebastian sschoop@tuev-nord.de

Reliable knowledge of the distribution of nuclide activities in a nuclear facility at the time of decommissioning forms the basis for decommissioning scenarios, dismantling and disposal studies and corresponding safety analyses. As an alternative to fine-meshed radiological sampling, the distribution of neutron-induced nuclide activities can be determined cost-effectively by simulating the neutron flux distribution and its effect on the structural materials of the reactor building. This quasi-continuous and realistic information offers the opportunity to reduce conservatism, which not only leads to cost savings through computer-aided optimization of segmentation and packaging, but also to an optimized workload (minimization through exposure times). In our presentation, we present modern calculation methods for corresponding simulations and discuss calculation methods and modelling decisions. In addition, we report on our experiences with the necessary preparations for high-quality modelling, i.e. the procurement of data such as geometry or material data of the structural materials. Furthermore, we show the procedure practiced at TÜV Nord for validating the calculation methods and models, with which the specified requirements for the quality of calculated values  can be shown.

505

## Updates on LILW disposal facility in Slovenia

Špela Mechora, Sandi Viršek

Agency for Radioactive Waste Management, Slovenia

spela.mechora@arao.si

Planed near-surface silo disposal facility for low and intermediate level waste (LILW) in Vrbina, Slovenia gained the environmental license in a year 2021 and early in year 2022 the Slovenian nuclear safety administration issued a consent for a construction. Construction license for a nuclear facility was issued in the year 2022 and for the infrastructure the construction license was gained early this year. The public procurement process for infrastructure and technical security was published at the end of last year and the contractor was selected. The tender for contractor for the LILW disposal container was successfully finished as well. The public procurement process for the nuclear facility construction is planned to be published in the first half on the year 2023 and the tender for the crane is planned to be published by the end of this year.

In the last few years extensive research of concrete mixtures for primary and secondary lining of the silo in the laboratory as well as on the field was done. Currently investigations for backfilling grout for the container are underway. First construction work is foreseen for the year 2023.

The article will describe the status of LILW disposal facility project in Slovenia and shortly present the disposal concept.

506

## Development of a wall crawling robot for remote inspection

Billy Murphy

University of Bristol, United Kingdom

billdorian@outlook.com

Working at heights poses substantial risks to workers' health, representing a leading cause of fatalities in the workplace and accounting for 24% of work-related fatal injuries in the UK. To address these hazards, innovative robotics systems are being developed to minimize the number of tasks performed at elevated positions. These robotic systems enhance operational efficiency and cost-effectiveness through their rapid deployability and elimination of conventional methods, such as cranes and scaffolding. British Magnox power plants are undertaking a rolling decommissioning programme estimated to cost up to £8.7 billion and will take up to 15 years before the sites will reach the "care and maintenance" stage. A specific area of interest for mitigating working at height is the inspection of boilers. These boilers when erected are 24 meters tall and sit within a compact boiler cell which when inspecting requires large and expensive scaffolding costing around £300,000 per boiler and taking months to construct whilst working in a radiological environment.

This paper presents a comprehensive account of the design and development of a pioneering wall crawling robot. The proposed robot uses permanent magnets for adhesion and incorporates advanced sensor technology, including Inertial Measurement Units (IMUs) and rotary encoders, to facilitate precise odometry. The platform is ideally suited to data acquisition from other instrumentation. Leveraging the obtained odometry data alongside radiation measurements from a gamma spectrometer, the robot enables accurate radiation mapping, thereby facilitating hotspot recognition and assessment.

507

## Characterisation of Uranium Metal Encapsulated in Magnox Sludge

Philip Hutchinson, Ross Springell, Tom Scott

University of Bristol, United Kingdom

vr19802@bristol.ac.uk

The UK has an aged inventory of spent uranium metal fuel that has become encapsulated in a magnox sludge during wet storage at Sellafield in the first generation magnox storage pond (FGMSP). The FGMSP has been used to store legacy magnox waste for decades; however, the contents of the pond must be processed and transferred to an alternative storage location so the FGMSP can be decommissioned. The encapsulation of uranium metal in the pond has occurred via the corrosion of the fuel rod cladding material, a magnesium alloy (magnox), that has formed thick layers of sludge [1]. Safe decommissioning of the FGMSP requires an understanding of any physical and chemical changes to the uranium metal due to storage in encapsulated conditions, as well as any other potential safety hazards the environment could pose.

The shift from a standard wet storage environment to an encapsulated environment, due to the presence of sludge, changes the possible corrosion reactions that uranium metal can undergo. The potential development of anoxic conditions, along with the ability of the sludge matrix to retain gas, promotes uranium metal corrosion via hydriding [2]. The formation and persistence of uranium hydride is important due to the generation and potential accumulation of flammable hydrogen gas, and the pyrophoric nature of uranium hydride when exposed to air [3].

In this work, uranium metal wire has been encapsulated in a magnox sludge to simulate the environment in the FGMSP. The corrosion behaviour of uranium metal has been monitored by comparing periodic x-ray tomography scans as the samples age. The scans have been used to extract quantitative data on corrosion rates at micrometre scale, as well as show morphological changes of both uranium, and magnox sludge.

[1] A. Laferrere, R. Burrows, C Glover, et al., “In situ imaging of corrosion processes in nuclear fuel cladding”, Corrosion Engineering, Science and Technology, vol 52:8, p. 596-604, 2017.
[2] A. Banos and T. B. Scott, “A review of the reaction rates of uranium corrosion in water,” Journal of Hazardous Materials, vol. 399, p. 122763, 2020.
[3] R. Orr, H. Godfrey, C. Broan, et al., “Kinetics of the reaction between water and uranium hydride prepared under conditions relevant to uranium storage”, Journal of Alloys and Compounds, vol.695, p.3727-3735, 2017.

508

## Recent Developments Regarding Fuel Services Technologies With Focus On Spent Fuel Management

Thomas Heinz Wiese[1], Jürgen Michna[1], Yannick Lotaut[2], Christophe Jarousse[2], Tony Pugh[3]

[1] Framatome GmbH, Germany

[2] Framatome SAS, France

[3] Framatome Inc., USA

thomas-heinz.wiese@framatome.com

Being a plant operator requires sustainable products and services for safe plant operation, especially related to fuel life time management. Framatome's Fuel Business Unit has knowledge, experience and skilled resources to meet the ongoing development of fuel assemblies (FA) and core components, as well as the evolving requirements for safe plant operation and decommissioning requiring customized solutions of detection, inspection and examination techniques. Our solutions consolidate decades of proven track record on on-site campaign performance and equipment supply with the ability to adapt to any evolving requirements of safety regulation or economical aspect. The presentation will exemplarily show developments of sipping techniques for FA with extreme long storage time; repair techniques of FA with focus on leaker detection but also specialized repair of FA structure components; sampling techniques for preparation of detailed hot cells examination; encapsulation of defective fuel rods for final disposal. Framatome's focus is on the technical aspects of the individual developed solutions especially for spent fuel management. You get an insight in sustainable developments in the above mentioned field of activities of the Framatome Field Services. The presented portfolio is the Framatome answer on the changing and evolving environment and based on successful projects executed currently or in the recent past. The wide range of the field services portfolio is fulfilling your requirements for new builds, for plant operation as well as for back end services for intermediate and long term storage.

509

## Uncertainty of Fuel Assembly Burnup in Siemens/KWU PWRs

Sebastian Schoop, Kai-Martin Haendel, Alexander Mühle

TÜV NORD EnSys GmbH & Co. KG, Germany

sschoop@tuev-nord.de

From a reactor physics point of view, fuel assembly burnup values are not limited by safety regulations. However, due to the correlation of safety-relevant parameters with burnup values, these values can represent a description of the licensing condition and are suitable for defining design limits. Accordingly, fuel burnup plays an important role in many aspects of fuel handling and storage. Although burnup itself is not a safety parameter, it serves as a state descriptor for a comprehensive characterization of the properties of a particular fuel type (e.g. uranium/MOX) and enrichment, e.g. in terms of reactivity, validity of safety analyses,

or as a starting point for decay heat or source term calculations. For licensing and safety analyses, it is usually necessary to use conservative burnup values. For TÜV NORD EnSys GmbH & Co. KG, as a technical expert organization that monitors the safe operation of nuclear facilities as a government contractor, knowledge of the burnup monitoring method and its uncertainties is therefore an important issue. This knowledge is needed to assess whether the safety margins from the safety analyses are still sufficient or whether another approach, e.g. BEPU approach, is necessary and valid.

An important factor in determining the fuel burnup is the power density distribution, so its uncertainties have a large impact on the uncertainty of the fuel assembly burnup. This presentation will give an overview of the method for determining the power density distribution in the Siemens/KWU PWRs of the1300 MW class based on the results of the Aeroball Measurement System, discuss the determination of uncertainties by comparing measured and calculated data and also present a selection of examples of the use of burnup data in safety analyses.

---

510

## The challenge of NORM waste from the production of titanium dioxide in Cinkarna Celje, d.d.

Nikolaja Podgoršek Selič

Cinkarna Celje, d.d., Slovenia

nikolaja.podgorsek-selic@cinkarna.si

The challenge of NORM waste from the production of titanium dioxide in Cinkarna Celje, d.d.

Cinkarna Celje is one of the largest Slovenian chemical companies. Its core business is titanium dioxide production. This is a white pigment used in a variety of products such as paints, plastics, paper, laminates, tiers, textiles, cosmetics, pharmaceuticals, etc.

One of the raw materials used to produce titanium dioxide is ilmenite. Ilmenite is found in the form of pebbles, and in the form of sand deposits. It also contains radioactive elements of natural origin, primarily uranium and thorium. Ilmenite itself does not belong to the category of radioactive materials that would require special handling.

When ilmenite is decomposed in sulphuric acid, sulphates of all the contained elements are formed from it. In the further production process, certain sulphates are excreted in the form of linings on the cold parts of the equipment. In this way, a gradual concentration of radioactive elements occurs, especially on rubber-lined steel equipment. When such equipment wears out and must be replaced, we are dealing with Naturally Occurring Radioactive Materials (NORM) waste.

In 1995, the health inspector issued a decision for the storage of NORM waste in a properly arranged warehouse in the company.

Due to the large volume occupied by the removed equipment, the contaminated rubber was removed from the steel parts and stored in drums.

In 2006, the company obtained a permit for the temporary storage of NORM waste.

For many years, the company has been looking for the possibility of permanent storage, processing into green concrete or incineration. No solution was feasible.

In 2014, Cinkarna obtained a Report on radiological measurements of contaminated material in temporary storage. The result was a record sheet with all the data for each individual stored drum. According to the measured specific activities and considering the Council Directive 2013/59/Euratom (EU BSS), most stored drums met the condition for waiving administrative control.

Cinkarna manages the non-hazardous waste disposal site Bukovžlak, which was slated for rehabilitation. We recognised the possibility of permanent storage NORM waste there and for this reason obtained an Assessment of the protection and exposure of the population due to the disposal of contaminated rubber and debris from the production of TiO2 at the Bukovžlak non-hazardous waste disposal site. The Environmental Agency rejected the possibility of installation because the landfill is closed, and it is no longer allowed to dispose waste there. Additionally, it is not allowed to dispose rubber.

We then looked for a solution in processing NORM waste into “green concrete” and using it by a remediation process at the Bukovžlak non-hazardous waste disposal site, but we received unofficial information that the Ministry of the Environment does not support disposal.

We began to investigate the possibility of incineration. First in Belgium at Belgoprocess, later at Studsvik in Sweden and finally Cinkarna made contact with the company Socodei in France, but the sulphur content was a problem for all of them, as well as the content of lead and other impurities in the rubber. They refused to burn it.

At the beginning of 2017, we received a negative response for the possibility of taking over and destroying the stored waste also from the authorised waste collector in Slovenia. Incineration in the cement industry was also not possible.

With the help of the Agency for Radwaste Management, Slovenia, we then got in touch with US Ecology, USA. After three years of checks, preparation of extensive documentation and the huge support of the Slovenian Radiation Protection Administration, we successfully exported 30 t of NORM waste and obtained a certificate for disposing in the state of Idaho.

With the successful completion of this project, Cinkarna eliminated the environmental and financial burden and freed up space for the development of technological processes that will maintain its operation for the future. Due to the changed procedure of treating contaminated equipment, this type of waste is no longer generated.

511

## Challenges and Objectives in Managing Institutional Radioactive Waste in Slovenia

Janez Vodopivec, Simona Sučić, Marko Kostanjevec

Agency for Radioactive Waste Management, Slovenia

simona.sucic@arao.si

Managing institutional radioactive waste poses numerous challenges, especially in countries with small nuclear programs where each waste stream tends to be unique. In such cases, providers of radioactive waste management have implemented diverse methods and developed various procedures for more common waste streams, which are now carried out routinely. The main goal of radioactive waste management is to prepare the waste in a suitable and safer form for long-term storage and eventual disposal. Another important objective is to reduce waste volume, which offers cost-effectiveness and benefits to operators with limited storage capacities and no immediate disposal options. Both aspects play a vital role in addressing institutional radioactive waste in Slovenia and during the past decades many measures were implemented addressing these two challenges.

The responsibility for managing institutional radioactive waste in Slovenia lies with the state public service. At the operational level, the main challenge for waste management operators is to identify appropriate methods for treating and conditioning radioactive waste, considering waste quantities generated in Slovenia, as well as human, economic, and infrastructure resources. This paper presents the methods, benefits, and experiences gained in the treatment and conditioning of radioactive waste, with a particular focus on implemented measures in reducing waste volume over the past decade, given the limited storage capacity at the Central Storage Facility (CSF).

512

## Characteristics of a Steel/Concrete Model System Under Repository Conditions

Margit Fabian, Istvan Tolnai, Otto Czompoly

Centre for Energy Research, Hungary

fabian.margit@ek-cer.hu

All states that engage in any kind of nuclear application must consider the management of radioactive waste and make sure it is handled in a safe manner, with due regard to the level of radioactivity and in compliance with national/international regulations. There is a broad consensus that the preferred method of ensuring long term safety for high level radioactive waste (HLW) is isolation in a deep geological repository (DGR), which will provide passive multibarrier isolation of radioactive materials. The vitrified HLW form in a steel canister is specifically designed for long term durability in storage and disposal. The requirements for lifetime and integrity of the steel canister depend on the DGR concept and geologic formation.

The focus of the interface scale is on two materials in contact to obtain information on the geochemical evolution close to an interface in terms of chemical variables and alteration in solid phase composition at a detailed small scale. The study concerns steel in contact with concrete with attention to the effect of degradation products from one material on the other. The concrete is originated from the Public Limited Company for Radioactive Waste Management (PURAM), considered as the buffer material in the final disposal program in Hungary.

We built a scale model system where close to real conditions were used (temperature, porewater composition). The main goal was to understand the characteristics, applicability and stability of the whole system, from the physical properties of the steel to the concrete response in the repository. We have been conducted a triplicate steel (S235JR carbon steel)/concrete (mixture of CEM II/B-S 42.5N) experiment using conditioned MQ-water for saturation. The corrosion potential was continuously monitored (container vs Pt electrode in clay). All containers were kept in an incubator at 80 °C. After 3, 7 and 12 months a container was opened for post-mortem characterization. With SEM/EDX investigations we focused on the composition and nature of alteration products formed on the steel and within the concrete. Formation of Fe-oxide ingrowths were detected and confirmed by micro-Raman investigations. The main corrosion products contain hematite and magnetite in different ratios. The soaking liquid was characterized using ICP-OES/IC after 3, 7 and 12 months. Higher K, B and Na concentration was found in the final water of the 7 and 12 month systems compared to the initial conditioned MQ-water. Elevated Ca and Na concentrations from baseline can be traced back to concrete content. The concentration of K and Si did not change significantly. Higher Mg concentration was measured after 12 months. The dissolution of all measured elements increased after 7 months to 12. The reactive transport code HYTEC was used to model the initial geochemistry of the concrete as well as its evolution with temperature and time. Details of the phase characteristics and trends of the dissolution rates will be presented.

The research activities presented were supported by the H2020 project EURAD-847593.

513

## Development of Verification Tool for the Review of Site Survey Result Obtained from MARSSIM Sign and WRS test

Jungjoon Lee

Korea Institute of Nuclear Safety, Republic of South Korea

lee@kins.re.kr

Recently, 2 commercial nuclear power reactors, Kori uni 1 and Wolsung unit 1, has been permanantly shutdown in Korea. Site characterization result could be included into final decommissioning plan which has been submitted for Kori unit 1, and will be submitted for Wolsung unit 1 by licensee (KHNP). Final site status survey plan is also suggested in FDP.

MARSSIM could be applied to entire process of radiation survey and site investigation from planning survey design to implementing final status survey (FSS) and confirmatory survey. Several procedure are provided in MARSSIM for checking whether the measured result in FSS can meet the release criteria. Two kinds of verification tool using MS Excel and R program are developed in order to conduct review of survey result obtained from MARSSIM Sign test and WRS test for every survey unit. These tool can make easily and earn time for the review process dealing with many kinds of data (measured data for survey unit and reference unit, DCGL, type I & II error, n, etc.) for statistical analysis.

514

## Deep Borehole Disposal as an Alternative Disposal Option for Spent Fuel

Leon Kegel, Sandi Viršek

Agency for Radioactive Waste Management, Slovenia

leon.kegel@arao.si

Many countries are developing a geological disposal project to dispose of their high-level radioactive waste (HLW) as well as spent nuclear fuel (SF) when considered as waste. The most widely selected option is the deep geological repository (DGR) concept, a mining repository located underground in a geological layer, in which conditioned waste is disposed of.

A potentially practicable, but less developed alternative to mined repositories that has received increased attention in recent years, is deep borehole disposal (DBD), involving disposal of highly active wastes in boreholes drilled to depths of a few kilometers for power programs or countries that do not use nuclear power but have small quantities of radioactive waste (RW) from other nuclear applications (such as research reactors) that also require geological disposal.

In 2021, ARAO investigated whether a deep borehole disposal could be used for disposal of SF from Slovenia's Triga Mark II research reactor. Deep borehole disposal for SF from Krško NPP was first analysed in 2022 through joint ERDO Association collaborative project. The project assessed the strategic potential of deep borehole disposal for several European countries, based on their existing and projected national waste inventories. DBD option of Krško NPP SF has been in 2023 additionally and more extensively analysed as part of the preparation of the Fourth Revision of the Krško NPP Radioactive Waste and Spent Fuel Disposal Program.

This paper provides basic design information and cost estimates about the deep borehole disposal for SF from Triga Mark II Research Reactor and SF from Krško NPP as well as conclusions and planned activities to further explore this disposal option.

515

# Direct conditioning of liquid organic radioactive waste into a geopolymer matrix

Anna Sears[1], Vojtech Galek[1], Thi Nhan Nguyen[2,3], Quoc Tri Phung[2]

[1] Research Centre Rez, Czech Republic;

[2] Belgian Nuclear Research Centre (SCK CEN), Belgium

[3] Department of Materials Engineering, KU Leuven, Belgium

anna.sears@cvrez.cz

The management and disposal of liquid organic radioactive waste pose significant challenges due to their hazardous nature and long-term environmental impacts. Conventional methods, such as incineration and solvent extraction, often result in secondary waste streams or generate harmful emissions. To address these issues, we investigate a novel approach for the direct conditioning of the liquid organic radioactive waste into a geopolymer matrix. A series of experimental tests were conducted to evaluate the feasibility and effectiveness of the direct conditioning method. Engine oil was used as a simulated organic radioactive waste and mixed with blast-furnace slag, alkali activators, and other additives to form a homogeneous geopolymer mixture. The mixture was then cured under controlled conditions, allowing the formation of a hardened geopolymer matrix incorporating the liquid organic radioactive waste. Various techniques, including UV/VIS, XRF, XRD, SEM, and compressive strength tests, were employed to characterize the physical, chemical, and mechanical properties of the resulting geopolymer waste forms. Compressive strength tests demonstrated that the geopolymer waste forms exhibited satisfactory mechanical performance, suggesting their potential suitability for long-term storage and disposal of liquid organic radioactive waste. Furthermore, leaching experiments were conducted to assess the leachability of oil and selected elements from the conditioned waste form. The results revealed very low oil leaching, indicating high immobilization efficiency. Overall, this study demonstrates the successful direct conditioning of liquid organic radioactive waste into a geopolymer matrix using blast-furnace slag alkali-activated binder. The results highlight the potential of this approach as a sustainable and effective solution for the immobilization and safe disposal of liquid organic radioactive waste streams. Further research and optimization are necessary to expand the application of this technique to other liquid organic waste types and evaluate its long-term performance under various environmental conditions.

516

## Caks and lifting equipment for remote handling of large highly radioactive parts of a spallation source

Szabina Török[1], Zsófia Kókai[2]

[1] Center for Energy Research, Hungary

[2] European Spallation Source, Sweden

torok.szabina@ek-cer.hu

The European Spallation Source, ESS, consists of a linear proton accelerator, a target monolith building with a tungsten target, neutron instruments and experimental laboratories. The accelerated protons impinge on the rotating target wheel and initiate spallation reactions, releasing neutrons from tungsten nuclei. During the operational time the structural materials will be activated. The tungsten target will have extreme high activity concentration due to the spallation process, as well as the proton and neutron activation. The target is planned to be replaced in every five years. Its activity concentration can be highly superior to parts of a utility plant e.g. a reactor vessel. In the qualitative radioactive hazard analysis different accident scenarios (e.g. cask drop) were studied that can happen during the cask operations.

For this reason, the high activity parts that have to be removed from the monolith for maintenance reason or for final deposition in the hot cell need to be manufactures with high precision to avoid leakage of gamma radiation is case of mal-functioning of the part.

The poster will show manufactured steel CASK containing even lead bricks that corresponds to the high dose reducing capability, complying with the enormous expectation.

517

## Improved Geopolymers For Encapsulation Of Molten Salts From Thermal Treatment Processes

Vojtěch Galek[1], Anna Sears[1], Petr Pražák[1], Martin Vacek[1], Jan Hadrava[1], Andrea Santi[2], Eros Mossini[2]

[1] Research centre Řež, Czech Republic

[2] Politecnico di Milano, Italy

vojtech.galek@cvrez.cz

Organic waste is commonly generated during the operation of nuclear facilities and decommissioning. High prices for repositories force Radioactive Waste (RaW) producers to lower the volume of organic waste for radioactivity content and chemical composition. Thermal treatment is one of the key routes for waste disposal and volume reduction, and Molten Salt Oxidation (MSO) is one of the potential processes. In the technology, the organic waste is dosed, together with oxidising medium, under the surface of the molten salt, where flameless oxidation takes place, and non-combustible materials are trapped in the molten salts.

For this study, the process for encapsulation of molten salt waste into an improved geopolymer matrix was determined. The commercial geopolymer LK was encapsulated with 10 wt.% of volcanic tuff as a filler. The volcanic tuff was chosen for zeolite richness for radionuclide trapping, good availability, and cost. The experiments were conducted with encapsulation of 5, 10, and 15 wt.% of waste salts into the matrix. The mixture was cured under controlled conditions, and its physical and mechanical properties were tested by SEM and XRD analysis. Compressive strength test demonstrated satisfactory mechanical performance for possible future use. Secondly, the molten salts were chemically enhanced for possible encapsulation. The molten salt waste is mainly comprised of $Na_2CO_3$ hydrates which can cause instability of the sample and cracking. The chemical reaction with calcium hydroxide was conducted to form a more stable $CaCO_3$. The resulting decantate had an average elemental composition of Na: Ca = 52.6:47.4. The resulting salt was encapsulated in a geopolymer with controlled conditions, and its physical and mechanical properties were tested by the SEM, XRF, and XRD analysis. The experiments were conducted with 5, 10, and 15 wt.% of enhanced waste added into the matrix. Compressive strength tests demonstrated satisfactory mechanical performance for future use, but more research is needed for possible waste load increase and sample stability.

# *Plant Life-Time Extensions, Reliability, Outage Management, Innovations and Modernization*

601

## Risk Informed Engineering Applications to reduce Operating and Maintenance Costs

Joseph Redmond

Framatome, United States of America

joseph.redmond@framatome.com

A Risk Informed Engineerng Program (RIEP) can enable nuclear plant owners to make technically sound decisions about plant design, operating and maintenance practices by using Probablistic Risk Analsyes and Operating experience. A RIEP can provide the technical bases for making changes to the NPP design, maintenance and inpsection programs that can contribute to safer and more cost-effective plant operation such as performing online maintenance, surveillance interval extensions, pricurement of spare parts, etc.
The process uses risk insights from the PRA and other sources to differentiate between components based on their importance to the overall safety of the plant or safety significance. Safety Related components determined to be low safety significance are categorized as a lower safety significance and exempted from the requirements of regulatory programs including, Nuclear QA program, Maintence Rule, Equpment Qualificaton, In Service Testing and Inspections.

602

## The MAI : An unique international collaborative research center dedicated to materials ageing for nuclear power plants

Marie Berthelot

EDF, France

marie.berthelot@edf.fr

The Materials Ageing Institute - MAI - is a utility-oriented research center founded and led by EDF and cofinanced by EPRI (US), KEPCO (J), CGNPC (CN), EDF Energy (UK), MHI (J), CRIEPI (J), CEA (F) and FRAMATOME (F). The main purpose of this collaborative effort is to bring together scientific skills and research facilities to address ageing of material used in nuclear power plants. This initiative by the world's biggest nuclear operators is motivated by the conviction that sharing research, experimental results, feedback and scientific information will significantly contribute to our understanding of the aging processes in various materials employed in nuclear power plants. It can then be used to anticipate ageing and henceforth increase the durability of material, components and structures in NPP.

603

## Mechanical Stress Improvement Process on NPP Krško Reactor Vessel Nozzles

Robert Planinc

Nuklearna elektrarna Krško d.o.o., Slovenia

robert.planinc@nek.si

Primary Water Stress Corrosion Cracking (PWSCC) became a nuclear industry issue in Pressurized Water Reactors (PWR) in year 2000 with through wall crack identified at Nuclear Power Plant V.C. Summer. PWSCC affects plants safety and economics. Crack to develop requires three conditions to be present: residual tension stresses on inner surface of piping wall as a result of welding; cracking susceptible welding material (like nickel alloy 82 or 182); and reactor coolant water chemistry. Extensive tension stresses and cracks are likely to be formed in bimetalic weld itself (weld between two different materials, usually stainless and carbon steel), axially and circumferentially, from inside towards outside of pipe wall. U.S. NRC responded upon this initial event with publishing several documents for mitigation and inspection of PWSCC in butt welds, which ultimately resulted in Code Case N-770 issued by ASME in 2005. The predominant methods used by the nuclear industry to mitigate and repair PWSCC susceptible welds are weld overlays (nickel alloy 52 or 152 applied) and the stress improvement. While main purpose of first is to improve structural condition of material, second actually improves stress state of the material.

Nuclear Power Plant Krško (NEK) is approaching to 40 years of commercial operation and entering life-time extension period. NEK adopted requirements of Code Case N-770 since it was adopted by NRC and is performing regular inspections of susceptible welds. In fact, NEK already mitigated majority of critical welds in the past in parallel with components replacements or simply by using some of mitigation methods. Even volumetric inspections on two hot legs, two cold legs and two safety injection (SI) reactor vessel nozzles to piping welds were showing no crack indications, NEK decided to act proactively. Based on many and positive experience with Westinghouse plants in U.S., NEK decided to implement stress improvement (MSIP) in outage 2022.

MSIP starts with extensive walkdown that consist of measurement and inspection of piping outside surface, inspection and measurement of primary system (RCS) piping supports and scanning MSIP tools layout area. Next phase is analytical; Ansys simulation of squeeze to determine location of MSIP tool and necessary permanent deformation of pipe for optimal stress transformation from tensile to compressive. Since radial pipe deformation results in pipe elongation, this side effect needs to be verified against existing piping and supports analyses. Ultimately steam generators and reactor coolant pumps translate as well, so verification includes their supporting system.

Following tool design and fabrication, MSIP is first simulated on a full-scale mock-up facility. Test squeeze is performed on hot leg, cold leg and safety injection pipe specimen. During mock-up training, all aspects of safety at work are also addressed.

MSIP was performed in October 2022. Since correct RCS piping supports adjustment is of utmost importance for plant safety, as found supports to piping/components gaps screening was performed immediately after plant shutdown, while still being at hot no-load conditions. During MSIP special attention is given to monitor pipe elongation and after MSIP proper supports gaps were restored. Nearby reactor vessel MSIP was performed with nuclear fuel assemblies outside reactor vessel and with coolant present in the piping to reduce radiological dose. MSIP operations were scheduled very thoroughly to reduce influence on outage overall duration as much as possible. Ultrasonic inspection performed after MSIP confirmed successful performance.

604

## On the reasons and extent of NEK's seismic instrumentation upgrade

Daniel Celarec[1], Milan Moškon[1], Alessandro Blascovich[2]

[1] Nuklearna elektrarna Krško d.o.o., Slovenija

[2] Geosig Ltd, Switzerland

daniel.celarec@nek.si

NPP Krško is designed so that, if the Safe Shutdown Earthquake (SSE) occurs, all structures, systems and components important to safety remain functional. In addition, all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake (OBE) in combination with normal operating loads. In NPP Krško, appropriate instrumentation is provided to monitor strong seismic ground motions at the site and seismic response of the NPP's structures so that the seismic response of the safety related structures, systems and components can be evaluated promptly after an earthquake. The function of strong motion seismic instrumentation system is to detect significant (strong motion) earthquakes at the site, record and manage (analyse and store) acceleration data, perform OBE and SSE exceedance evaluation, and report to plant operator during and after an event. During the operation of the NPP Krško, the strong seismic motion monitoring instrumentation was upgraded three times. The last and the most extensive upgrade was performed in 2022, when the system was completely replaced by new one. New system was extended with several new ground motion sensors locations on free field and inside safety related buildings in order to ensure compliance with up-to-date regulatory guidelines for the industry (i.e., both new Bunkered Buildings, new Dry Storage Building, and new locations at Main Nuclear Island). The free-field sensors close to the Main Island (sensor A1) was relocated to being immediately adjacent to another free-field sensor (A2) in order to avoid signal disturbances during earthquake due to effect of seismic response of nearby structures (at close vicinity) on recorded ground motion at the location of sensor A1. The set of free field sensors is completed by the down-hole sensor A9, which has been installed following the requirements of the latest regulatory guide (RG 1.12 rev. 3). Obsolescence and operational and maintenance issues were also a very important reasons for the old system replacement. Specifically, the old system was occasionally subject to false triggering due to impacts of electric discharges on the outdoor signal and triggering lines during lightning strikes, and negative effects of electro-magnetic radiation on in-housed lines. Another deficiency was exposure to effects of radiation and high temperature that could have caused system malfunctions due to the failure of the recorders in the reactor and intermediate building, respectively. In the new system: (1) optical connections between the outdoor recorders in main control cabinet in the Main Nuclear Island are used to avoid lightning impacts; (2) electrical lines between the sensors and recorders are designed and installed following up-to-date electro-magnetic compatibility qualification code requirements to avoid disturbing electro-magnetic effects; and (3) all in-housed recorders are relocated from the sensor locations into the main control cabinet. The new cabinet was expanded to include all recorders in-housed inside Main Island in order to avoid negative radiation and environmental effects on the recorders. Entire system was seismically qualified. Installation of the new system began in mid-2022 with the installation of cable trays, temporary relocation of free field sensors, installation of new free field sensors and a downhole sensor, and was finally completed with the installation of a new cabinet in the main control room, testing and commissioning during the outage the same year.

605

## Enhanced Indoor Inspection of Nuclear Facilities through Non-Rigid Airship-Based SLAM System

Shanqi Song, David Megson-Smith, Matthew Ryan Tucker, Yannick Verbelen, Tom Scott

University of Bristol, United Kingdom

nm19805@bristol.ac.uk

This paper presents a novel approach for indoor inspection of nuclear sites using a non-rigid airship (NRA) equipped with the RPLIDAR S1, BerryIMU v4 and Raspberry Pi for simultaneous localization and mapping (SLAM). The proposed system addresses the limitations of multi-rotor UAVs, including the particulate aerosols caused by downdraft, limited flight time and radiation affecting onboard electronics.

The NRA's utilization of helium gas enables extended weeks of stationary flight due to buoyancy, surpassing the flight time constraints of multi-rotor UAVs. Furthermore, the simpler onboard electronics of the NRA are less susceptible to radiation, eliminating the need for complex shielding. Additionally, the gentle upward draft created by the NRA reduces disturbance to the environment, minimizing aerosolization compared to the downdraft generated by multi-rotor UAVs.

The proposed system incorporates the RPLIDAR S1 2D laser scanner and BerryIMU v4 for localization and mapping. To obtain distance data from the third dimension, a mirror is attached to the side of the RPLIDAR at a 45-degree angle, transforming the 2D lidar into a 2.5D lidar. This modification enables the RPLIDAR to capture both vertical and horizontal information. The Raspberry Pi serves as an onboard computer, running ROS nodes for the RPLIDAR, and transmitting data to a laptop for further processing. This multi-sensor integration meets the limited payload requirement of the NRA (~200g for a 2m long, 1.5m^3 NRA), ensuring efficient operation within the system's constraints.

The proposed approach offers a safer and more efficient method of inspecting nuclear sites indoors while minimizing risks to personnel and equipment. By employing the NRA for indoor inspections, significant cost and complexity reductions can be achieved while enhancing the accuracy and comprehensiveness of data collection.

606

## RPV threaded hole lubrication device

Domen Zorko, Matej Pleterski, Aleš Jevnik

Numip d. o. o., Slovenia

matej.pleterski@numip.si

Adequate lubrication of RPV stud bolts is crucial for regular RPV head area maintenance. A newly developed specialized device for lubricating RPV threaded holes is presented. The device is designed in order to allow adjustments of all variables, i.e. travel and speed of stroke, lubricant supply and spray pressure. Besides that it comprises also plug seat surface sealing and constant lubricant stirring and circulation function. It is lightweight and operates using compressed air only. The device offers automatic lubrication cycles, high productivity, repeatability and reliability while also remaining ergonomic and user friendly.

607

## LT Cam : a contactless surface inspection tool with high sensitivity

Severine Annaval

Intercontrole, France

severine.annaval@framatome.com

Framatome, Intercontrole and Edevis has developed a new imaging, non-contact Non-Destructive Testing tool which can be used as an alternative to penetrant testing and magnetic particle testing for the detection of emerging or underlying defects.

LTcam is an industrial inspection system using infrared thermography in combination with Laser heating. Active Thermography (TT) is a non-destructive inspection method using heat flux as a mechanism to detect flaws. LTcam is a contactless device based on the detection of infrared (IR) emission generated by a thermal Laser excitation. It consists of a high power Laser source, an optical system allowing the transformation of the Laser beam into a Laser line, an infrared camera and a 2D scanner to scan the surface of the inspected part.

The surface to be inspected is locally heated with a focused Laser line. The infrared emission of the surface near the heating line is measured by an infrared detector. A scanning of the heat flux on the part to be inspected allows defect detection by the thermal barrier effect induced by these defects. When the

excitation/detection (E/D) system is approaching an opening or an underlying flaw, the infrared signal increases because the heat is no longer distributed in the direction of scanning. When the (E/D) system passes over the defect, the detected signal decreases because the defect prevents the heat generated by the Laser from diffusing into the detection of scanning. Continuous measurement of the infrared signal during the scanning and reconstruction of the information into a 2-D representation makes it possible to display a complete image of the inspection surface. The IR image can be used to manually or automatically detect flaws. The operator is able to analyze the resulting IR images with different ROIs, where the thermal signal characteristics are evaluated.

Customer benefits for LTcam using:

- Modular, adaptable and scalable system. Automatic, semi-automatic or manual inspections.
- All types of materials and surface states. Inspection without any contact with the piece
- Can be used in laboratory or in production
- Inspection rates up to 2 m²/hour
- Detection of small opened and sub-surface flaws (less than 1mm length; few microns opening)
- The inspection of zones difficult to access is possible with LTcam (cavities, drillings, tubes.) thanks to an additional tooling system with mirrors
- Tested on different materials (ferritic and stainless steels, Inconel, aluminum, …) and on different surface states (raw, grinded, machined, polished…)
- Inspection of huge surfaces without moving the camera (up to 2 m²)
- Possible help for automatic diagnosis
- Automatic generation of inspection report and storage of digital data
- Built in accordance with the EU regulation REACH
- Built in accordance with the security standards (Laser, electric and machinery)

608

## Beznau RPV Shell UT Inspection and Indications Follow Up

Severine Annaval

Intercontrole, France

severine.annaval@framatome.com

Consecutively to the Hydrogen flakes detected in the base metal of DOEL 3 and TIHANGE 2 RPV core shells in 2012 and 2013, AXPO utility has performed an inspection of the same areas for Unit 1 and 2 RPV of BEZNAU NPP. After detection of some indications, the utility asked to deploy the UT Inspection technique (immersion technique with focused probes) already qualified for Belgium plants to inspect the suspected area. As the INTERCONTROLE MIS machine was not available for this inspection, the CMM (a Framatome Germany manipulator) has been adapted to implement the qualified UT inspection technique. To achieve and valid this solution the following steps were:

- Design and manufacture the necessary adaptation of the CMM manipulator to adapt the UT equipment system of INTERCONTROLE,
- Validation of the manipulator configuration thanks to trials in Hot workshop,

- Qualify the associated UT inspection procedure,
- Performance of the inspection in 2016.

Considering the results of the initial inspection, the Swiss Safety Authority has required to follow a sample of the indications recorded in 2016 by a second inspection in 2022 in order to warranty any indication evolution. It is mandatory that the same equipment and the same UT technique have to be used in order to warranty an accurate and fair comparison. For this purpose, it is necessary to establish the comparison criteria of the UT technique in order to distinguish an indication evolution from a deviation coming from the general UT process (acquisition and analysis). As it was a major milestone for the continuous service of the unit till its end of life, the following steps have been taken previously to the inspection:

- Trials on a representative mock-up containing real defects in order to evaluate the influence of the essential parameters (water path, orientation of the probe, repeatability and reproducibility, effect of the probe, water temperature) on the indication detection and characterization. The results of these trials have been used to define the criteria for indication comparison between two inspections. A new UT inspection procedure has been written to define the comparison process to apply between two successive inspections,
- Validation of the manipulator configuration thanks to trials in hot workshop,
- Performance of the inspection in 2022.

The second inspection results showed that no evolution on the whole sample of indications to be assessed. Prior to the inspection, the trials on a representative mock-up were necessary to justify the comparison criteria. Face to the challenging evaluation of indications comparison, the field feed-back of this BEZNAU experience shows that the common inspection teams of Intercontrôle and IB-G has demonstrated its reliability and flexibility to address such difficult process. Our proposed solutions have been effective and fully met the utility’s expectations.

---

609

## Completion of the Fourth 10-year ISI Program with Emphasis on Reactor Vessel Condition

Justin Božič, Krešimir Gudek

Nuklearna elektrarna Krško d.o.o., Slovenija

justin.bozic@nek.si

NPP Krško Inservice Inspection Program (ISI Program) TD-2H/4 describes the purpose, objectives, responsibilities, requirements, program plans, and related drawings for the 4th inspection interval that started in July 2012 and ended in July 2022. ISI Program TD-2E/4 is also credited for management of aging effects in accordance with requirements defined in Aging Management Program (AMP).

NPP Krško Technical Specification SR 3.0.5 "Inservice Inspection and Testing of ASME Code Class 1, 2, and 3 Components" requires that Inservice inspection shall be performed in accordance with ASME Section XI as required by 10 CRF 50.55a(g), except where relief has been granted by the URSJV pursuant to 10 CFR 50.55a(g)(6)(i).

10 CFR 50.55a determines in paragraph (b) applicable Edition (and Addenda) of ASME Section XI. Paragraph (g) determines relation between the Edition of Section XI incorporated by reference in paragraph (b) and inspection intervals of ISI Program. ASME Section XI provides rules for the examination, testing and inspection of Class 1, 2, and 3 components and systems. US NRC Regulatory Guides may require an additional examination when the component part is not covered by the ASME Section XI. The inservice examinations and system pressure tests required for Class 1, 2, and 3 components and systems shall be completed during each of the inspection intervals for the service lifetime of the power unit. ASME Section XI, Division 1 has been mandatory for NPP Krško from the beginning of operation. It defines scope for class 1, 2 and 3 of the pressure retaining components and their supports as well as that plant owner is responsible for the ISI Program. ISI Program shall define exam methods, scope and frequencies of examinations, acceptance standards, system pressure tests, records and reports and repair/replacement activities.

NPP Krško ISI Program for the 4th inspection interval is developed based on risk informed (RI) principles in accordance with Electric Power Research Institute (EPRI) Technical Report TR-112657 Rev. B-A “Revised Risk Informed Inservice Inspection Evaluation Procedure”. It is also conducted in a manner consistent with ASME Code Case N-578-1 “Risk-Informed Requirements for Class 1, 2 and 3 Piping, Method B” and ASME Section XI Nonmandatory Appendix R, “Risk-Informed Inspection Requirements for Piping”. The objective of the RI ISI Program is to identify risk-important piping segments, to define the elements (welds) within this risk important piping that are to be inspected, and to identify appropriate inspection methods for these welds. Risk-important pipe segments are identified by the consequence evaluation, which focuses on the impact of a postulated pipe failures in a consistent manner, based on their impact on plant safety.

After RI-ISI Program has been prepared it passed through process of acceptance or approval. This includes independent review and approval by Slovenian Nuclear Safety Agency through “Relief Request” as required by 10CFR50.55a(a)(3)(i).

In addition to the elements selected by the RI application, NPP Krško ISI Program included augmented inspections related to the Nickle-Based Alloy 600/82/182. Additional inspections came from requirements of Code Case N-722-1 (Augmented examination of reactor coolant pressure boundary components), Code Case N-770-2 (Augmented examination of Class 1 piping and nozzle dissimilar-metal butt welds) and Code Case N-729-4 (augmented examination of PWR reactor vessel upper heads with nozzles having Pressure-Retaining Partial-Penetration Welds).

NPP Krško reactor pressure vessel has been manufactured by ASME Sec. III and pre-service inspection (PSI) performed in 1979. In each successive 10-year inspection interval, one full and one or two partial scopes of RPV welds UT inspection were performed. In 2010 full scope of UT inspection was performed in accordance with provisions of the ASME Sec. XI, Appendix VIII (Performance Demonstration) using conventional UT technique. 13 flaw indications were detected and evaluated as acceptable. All flaws are characterized as results of welding process – metallurgical origin and are not cracks. In 2021 at the end of 4th inspection interval, same UT inspection scope was performed using new developed UT technique – “Phased Array”. This inspection provides more information and allows more precisely evaluation. The indication comparison showed no flaw propagation for indications previously reported.

In response to the EPRI Material Research Program MRP-227, in outages 2021 and 2022 NPP Krško performed inspections to fulfill commitments from Aging Management Program, which is developed in accordance with NUREG-1801, Rev. 2, GALL; XI.M16A PWR Vessel Internals. Visual inspection was focused on control rod guide tube assembly guide cards, control rod guide lower flange welds, core barrel upper flange weld, core barrel girth weld, thermal shield flexures, lower internals edge bolts and baffle-former assembly. Baffle-former bolts were inspected by ultrasonic testing.

All examinations performed in the scope of the ISI Program for the 4th interval were performed by certified examination personnel and in accordance with approved and qualified procedures. Scope of examinations was fulfilled 100%.

As a general conclusion, NPP Krško primary system components in the scope of the ISI Program are in good condition for the safe operation for the next 20 years.

610

## Implementation of the NEK PSR3 Project and Action Plan

Rudolf Prosen[1], Špalj Srđan[1], Barbara Colo Vrdoljak[2]

[1] Nuklearna elektrarna Krško d.o.o., Slovenija

[2] Enconet, Croatia

rudolf.prosen@nek.si

The Periodic Safety Review (PSR) is a comprehensive safety review of all important aspects of safety, carried out at regular intervals, typically every ten years. Within PSR, the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects are assessed. A PSR includes an assessment of plant design and operation against applicable current safety standards and operating practices, and has the objective of ensuring a high level of safety throughout the plant's operating lifetime. The PSR project for Nuclear Power Plant Krško has been initiated per the requirements of national Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) and the regular performance of PSR in 10-year intervals is a prerequisite for plant lifetime extension to 60 years.

NPP Krško (NEK) undertook the 1st PSR, performed in 2001, 2nd PSR in 2010 and the 3rd PSR, which is ongoing from 2020. The third PSR is one of three supporting projects for the Lifetime Extension. The article will present the results of the review as available at the time of writing and also implementation of the experiences from the second PSR review, the good practices and weakness resulted to the approach taken.

611

# Robotic non-destructive inspection of corrosion inside steel cylinder concrete pipes used in nuclear power plants

Nina Gartner[1], Miha Hren[1], Andraž Legat[1], Sokratis Iliopoulos[2], Wouter Van Eesbeeck[2], Lucas Si Larbi[3], Eric Lucet[3]

[1] Slovenian National Building and Civil Engineering Institute, Slovenia

[2] Belgisch Laboratorium van de Elektriciteitsindustrie Laborelec CVBA, ENGIE Laborelec, Belgiu

[3] List, CEA, Université Paris-Saclay, France

nina.gartner@zag.si

The EU-funded (H2020) ACES project (https://aces-h2020.eu/) investigates ways of advancing the current assessment of safety performance for the long-term operation of concrete structures in nuclear power plants (NPPs). Concrete structures with embedded steel liners are a significant part of the NPP's infrastructure. While concrete enables passivation of the steel surface due to its alkalinity, corrosion processes can still initiate due to environmental conditions (e.g., chlorides) and steel/concrete interface geometry (e.g., a crevice). Many of these corrosion processes remain hidden inside the concrete until corrosion propagates to the stage where mitigation becomes complex and costly. One of the main corrosion concerns in NPPs is the chloride-induced corrosion of steel cylinder concrete pipes (SCCPs) used in cooling water systems.

SCCPs are composed of three layers: an inner concrete cover (with or without steel reinforcement mesh), a steel plate, and an outside concrete cover (with or without steel reinforcement mesh). The ACES project focuses on finding suitable non-destructive testing techniques (NDTs) that enable early detection of steel liner corrosion inside the pipes. Moreover, a mobile robotic platform with a manipulator was specially developed within the ACES project to automate the process of manipulation of selected corrosion NDTs inside SCCPs.

The evaluation and selection of corrosion inspection NDTs was performed in two steps. Firstly, mock-up specimens were fabricated for the initial laboratory study of potentially appropriate NDTs. The mock-ups were designed to closely mimic SCCPs and enable controlled evaluation of NDT abilities, such as the detection of artificial defects, influence of different concentrations of mixed-in chlorides and different concrete cover thicknesses. Several electrochemical and physical NDTs were evaluated in a laboratory environment, and two of them were selected as the most promising: Pulsed Eddy Current (PEC) and half-cell Potential Mapping (PM). PEC is an electromagnetic inspection method used to detect steel thickness loss in ferromagnetic objects, while PM measures the potential of steel embedded in concrete relative to a reference electrode on the concrete surface, indicating the likelihood of corrosion. Both selected NDTs are commercially available, cost-effective, portable, robust, and relatively easy to use. Subsequently, both selected techniques were manually tested inside two types of SCCP already used in nuclear power plants. One of the two SCCPs showed no visible signs of corrosion damage, while the other (with additional inner steel mesh) was severely damaged. The selected NDTs were found to be complementary, as PM could indicate the likelihood of corrosion even at low corrosion activities, while PEC could confirm and quantify the severity of corrosion damage. Based on the results obtained from laboratory mock-up specimens and real SCCPs, it was concluded that both tested NDTs for corrosion measurements provide useful information and are suitable for further testing with the robotic platform.

Specific environmental and geometric considerations were taken into account when developing the robotic solution for use inside SCCPs. A wheeled mobile platform was selected as the most suitable option. In addition to the two selected corrosion inspection NDTs, the platform is equipped with additional hardware that allows smooth navigation and comprehensive corrosion inspection. These include a remote communication system, inertial measurement unit, 2D lidar and stereo camera for navigation, outdoor PTZ and panoramic cameras for remote monitoring. The rotating and deploying robotic arm was custom-designed for the manipulation of specific rod electrodes for corrosion inspection inside SCCPs.

The next step in the ACES robotic corrosion inspection solution will involve conducting a case study demonstration. Corrosion measurements will be performed inside an old SCCP with selected NDTs installed and operated on a mobile robotic manipulator. The aim is to assess the implementation of an automated measurement protocol specific to each NDT, and to compare the results obtained with manual measurements. This final step will take place at the CEA premises in Paris/Saclay in August 2023.

---

612

## The Assessment of Dose Rates during MPC Loading and Drying in frame of the Nuclear Power Plant Krško First SFDS Loading Campaign

Davor Grgić, Paulina Dučkić, Mario Matijević, Siniša Šadek

University of Zagreb Faculty of Electrical Engineering and Computing, Croatia

paulina.duckic@fer.hr

In this paper, dose rates around HI-TRAC cask will be assessed at specific Fuel Handling Building (FHB) locations where plant personnel is expected to spend some time during activities related to Multi Purpose Container (MPC) loading and drying as a part of the first Nuclear Power Plant Krško (NEK) Spent Fuel Dry Storage (SFDS) loading campaign. Considered FHB locations are Cask Loading Area (CLA), Decontamination Area (DA) and characteristic locations on a transfer path between them. In CLA, wet configuration (SFP water in MPC) and in DA both wet and dry (helium in MPC) configurations are considered. The actual neutron and gamma sources for real casks are used in hybrid shielding MCNP6/ADVANTG calculation. The main goal of this work is to use the obtained results in estimation of typical doses acquired by plant personnel during the most demanding spent fuel transfer activities in NEK and compare them to available measurements.

613

## Early Launch of Validation via an Evolving Engineering Simulator (ELVEES)

Peter Hruščak, Guillaume Hémery

Framatome, Slovak Republic

peter.hruscak@framatome.com

In designing a new nuclear power plant, a full-scope simulator (FSS) is developed alongside and is necessary for the training of plant operators.

It is also often used as supporting tool when performing integrated system validation before the plant is commissioned.

Notwithstanding the significance of the FSS role, it typically becomes operational too late for both validation activities and the resolution of any findings in a manner that allows complete control of the project schedule.

That's why we explored whether a lower-fidelity FSS, accessible to the design team during the project's design phases, would enable and support the early and iterative validation of engineered HMI displays and other I&C systems.

This lower-fidelity simulator known as ELVEES will evolve in parallel with the progress of the project and the increasing maturity of the deliverables' design.

614

## Modernizing Manual Ultrasound NDE Inspection of Pipes in Nuclear Power Plants with Machine Learning

Željko Rapljenović, Luka Posilović, Marko Budimir

INETEC - Institute for nuclear technology Ltd., Croatia

marko.budimir@inetec.hr

In the nuclear power industry, the safety and reliability of piping systems are critical to ensure the safety of the plant and the environment. Manual ultrasound Non-Destructive Evaluation (NDE) techniques have traditionally been used to detect defects such as cracks, corrosion, and leaks in these systems. However, these techniques can be time-consuming, and the results are subject to interpretation by the inspector.

To address these challenges, we propose a new conceptual solution that integrates machine learning algorithms into the ultrasound inspection instrument to assist inspectors in real-time defect detection. Our solution focuses on manual NDE inspections, which are the key to achieving high levels of performance in flaw detection reliability, flaw sizing accuracy, and low false call rates. The algorithms are planned to be built-in into the ultrasound inspection instrument, and a machine learning-based classification and sizing algorithm is included.

Additionally, our solution includes the Performance Demonstration Initiative (PDI), which provides a standardized approach for verifying the performance of NDE equipment and personnel. The PDI inspection is used to demonstrate that NDE procedures, equipment, and personnel are capable of achieving a specified level of performance.

We have conducted initial experimental studies to evaluate the performance of our proposed solution, and the results have demonstrated its efficacy in detecting defects in real time. We believe this has the potential to modernize the inspection of piping systems in nuclear power plants.

# *PSA and Severe Accidents*

701

# Fuel Cladding Deformation during LOCA: Comparison between Single-rod and Multi-rod tests

Cristina Dominguez, Tatiana Taurines, Georges Repetto, Emmanuel Rouge

IRSN, France

christina.dominguez@irsn.fr

The purpose of the COCAGNE experiments, performed in the framework of the PERFROI program, is to study the deformation and burst of fuel cladding under Loss-Of-Coolant-Accident (LOCA) conditions in "multi-rod" configuration. This configuration enables to study the deformation in situations of contact/blockage between peripheral rods, the burst conditions after contact, and the influence of azimuthal temperature gradient on cladding strains. The main objective is to get precise knowledge of the three-dimensional mechanical behaviour of pressurized rods surrounded by structures simulating four neighbouring fuel rods, under a geometrical and thermal representative environment.

Six ramp tests have been performed in single-rod configuration and seventeen in multi-rod configuration (configuration taking into account the contact with peripheral rods). The tests were performed on single-sided pre-oxidized claddings (10 µm of external pre-oxide) or double-sided pre-oxidized claddings (10 µm of external and internal pre-oxides) at three heating rates (1, 5 and 10 °C/s) and at pressures from 2 to 10 MPa. The effect of the temperature gaps between the test rod and the guards or between the guards were also studied.

The cladding tube and the guards are all heated electrically by direct current. Tests are performed under vacuum atmosphere. The temperature and the radial deformation of the external surface of test rod are measured on-line along three parallel generating lines of the cladding.

Burst strain and axial propagation of ballooning measured after the multi-rod tests were in most cases lower than those expected. Various factors have been identified as responsible of the reduction of ballooning, as the presence of an internal pre-oxidize layer in the cladding, temperature gradients (induced by the guards or generated by the misalignment of the pellets inside the rod) or the limitation of azimuthal deformation by the presence of the guards that simulate peripheral deformed rods.

Two kinds of apertures are formed on burst balloons: the first one (observed in tests performed at 2 and 5 MPa, in which burst takes place in the $\alpha+\beta$ region) has elongated fish-mouth shape; the second one (observed in tests performed at 10 MPa, where burst takes place in the $\alpha$ region) has a rectangular shape. The rectangular aperture is probably caused by the violence of burst at 10 MPa.

There is a linear dependence between the dimensions of the fish-mouth shape apertures formed in the test performed at 5 and 10°C/s. In the tests performed at 1°C/s, burst strain is higher and the aperture is constrained by the presence of the guards.

The presence of the four guards affects the size and the shape of the formed balloon. In multi-rod tests, azimuthal deformation is limited by the guards leading to lower maximal strains compared with single-rod tests. Nevertheless, under certain boundary conditions, the ballooning can develop in the axial direction. The round balloon formed in single-rod tests, adapt to the available space between the guards in multi-rod tests becoming squarer.

Experimental data are currently used to validate three-dimensional thermal mechanical models implemented in the DRACCAR software (developed by IRSN for the simulation of LOCA type accidents) that can simulate rod and multi-rods behaviour during LOCA transients.

702

## Development of 3D fracture mechanics submodels for PTS analyses of an RPV with cracks

Oriol Costa Garrido, Nejc Kromar, Andrej Prošek, Leon Cizelj

Institut "Jožef Stefan", Slovenia

oriol.costa@ijs.si

Pressurized thermal shock (PTS) analyses are required to assure that, during the operation of a pressurized water reactor, potentially existing crack-like flaws in the reactor pressure vessel (RPV) wall will not initiate and propagate during loss-of-coolant-accidents (LOCAs) or other PTS-relevant transient scenarios. A finite-element model (FEM) of the RPV is employed in a thermo-mechanical analysis to evaluate the temperatures and stresses in the RPV wall during the transient. For models that include a postulated crack, stress intensity factors (SIFs) are obtained together with other results in a fracture-mechanics analysis. The inconvenience of the latter is that different models of the entire RPV are required to study different crack shapes, sizes or orientations. Additionally to the fact that meshing of the crack is a rather complex task in itself, consideration of the relatively small crack typically also complicates the meshes and increases the overall number of elements in the entire RPV model. To avoid this, the submodeling technique is used, where the thermo-mechanical analysis is performed with the FEM model of the entire RPV without cracks, and submodels of a small portion of the RPV containing the cracks are used in separate fracture-mechanical analyses. The displacements obtained in the thermo-mechanical analysis at the submodel boundary surfaces are employed as boundary conditions.

This paper presents the development of fracture-mechanics submodels in Abaqus containing axially and circumferentially oriented through-clad cracks (TCC) and under-clad cracks (UCC). A previously developed three-dimensional (3D) FEM of an RPV with four cooling loops is employed in the thermo-mechanical analysis of a small-break LOCA (SB-LOCA). The goal of the paper is to develop the submodels and their meshes to accurately analyze SIFs of the cracked RPV for future PTS analyses. The SIFs will be compared with the results obtained using existing semi-analytical formulae and with the FAVOR code. This work has been performed in partial fulfillment of the European project APAL (Advanced PTS Analysis for LTO).

703

# The Scaling of Turbulent Flame Acceleration and Detonation Transition for Hydrogen-Air mixtures in the RUT Facility

Mike Kuznetsov, Alexei Kotchourko, Wolfgang Breitung

Karlsruhe Institute of Technology, Germany

kuznetsov@kit.edu

To model the Loss of Coolant Accident (LOCA) in a containment of nuclear power plant (NPP) a series of large scale bench-mark experiments have been conducted in the RUT facility. Flame propagation regimes for very lean hydrogen-air mixtures from 10 to 14% H2 have been investigated in the RUT set-up. The facility consists of three parts: the first channel (34.4 x 2.5 x 2.2 meters), the canyon (10.5 x 2.5 x 2.2 m ) and the second channel (20.1 x 2.5 x 2.2 m) Total volume of mixture was about 480 m3. 30 and 60% of the channel cross-section was blocked by concrete blocks. Slow and sonic deflagrations in the channel have been established for hydrogen compositions up to 12.5%H2 as well as a detonation transition at 14% of hydrogen was registered. In a canyon of a bigger dimension, the detonation was observed even at 12.5%H2 due to shock reflection at the far corner of the canyon. Such detonable concentrations for channel and canyon geometries significantly extend the well-known conventional detonability limits of 18-59% H2 (NASA STD 8719.16 - Safety Standards) and demonstrate the very high danger of the detonation even for very lean hydrogen-air mixtures.

Since the critical conditions for DDT are usually analyzed in terms of the dimensionless ratio of the characteristic size of the system to detonation cell size l of the mixture (as a measure of mixture sensitivity to detonation initiation), the scaling down of flame acceleration and detonation transition was investigated for similar geometry in a MINIRUT facility of 50 times smaller size (1:50) than the original RUT facility. The advantage of the small-scale facility was that it allows high-speed video monitoring of flame acceleration and detonation transition. The detonation in the channel and in the canyon of 50 times smaller scale has occurred for hydrogen-air mixtures with a detonation cell size of 50-57 times smaller than for a large-scale RUT facility. The known criterion 7λ for detonation onset in an obstructed channel was experimentally confirmed. The video monitoring confirmed the shock reflection and shock-flame interaction mechanisms for deflagration to detonation transition (DDT). Beyond the detonation cases for leaner hydrogen-air mixtures, the flame can propagate as subsonic or sonic deflagration. The criterion for sonic deflagration will be the critical expansion ratio σ = 3.75 which corresponds to 10.5-11%H2. The experiments also confirmed that the flame dynamic for deflagration mode does not scale down as the DDT process. Even more, below σ = 3.75 the flame can propagate faster in narrower channels at the same hydrogen concentration.

704

## Simulation of an SBO accident in a PWR with AC2/ATHLET-CD with and without using ATF cladding material

Liviusz Lovasz, Peter Pandazis

GRS, Germany

liviusz.lovasz@grs.de

ATHLET-CD (Analysis of THermal-hydraulics of LEaks and Transients with Core Degradation), which is part of the system code package AC2 (ATHLET, ATHLET-CD, COCOSYS) is designed to simulate the behaviour of a nuclear power plant during accident scenarios, including core damage progression as well as fission product and aerosol behaviour and lower plenum phenomena. The code system is being developed and improved by GRS. Previously, models and input decks were using standard, zirconium-based materials as cladding. These cladding materials have been used in nuclear reactors all over the world, despite their exothermic reaction with hot steam that can aggravate accident progression and produce easily combustible hydrogen. In order to eliminate or mitigate this negative characteristic of the zirconium-based claddings new materials were developed. These so-called accident tolerant fuel/cladding (ATF/ATC) materials have much better oxidation characteristics during some or most accident conditions.

705

## Seismic risk assessment of safety-related SSCs

Matjaž Dolšek

University of Ljubljana, Faculty of Civil and Geodetic Engineering, Slovenia

mdolsek@fgg.uni-lj.si

Seismic risk assessment of safety-related SSCs by rigorous methods of seismic analysis is very complex and computationally demanding. As a consequence, simplified seismic risk assessment methods are often used. In the paper, a brief overview of the seismic risk assessment methods of safety-related SSCs is presented. Focus is on the hazard-consistent selection of seismic actions for seismic fragility analysis, the uncertainties affecting the seismic fragility analysis and the uncertainties in the risk integral that are related to the lower and the upper bound seismic intensity considered in the risk integral. The issues of the limit states for which the fragility functions are assessed are also discussed. The second part of the paper discusses how the seismic risk assessment can be considered in the design process aimed at upgrading a critical component to reduce the overall risk.

706

# Probabilistic Risk Assessment is Conservative or Best-Estimate Approach?

Seyed Ali Hosseini, Francesco D'Auria

University of Pisa, Nuclear Research Group of San Piero a Grado (GRNSPG), Italy

ali.h.research@gmail.com

The Deterministic Safety Assessment (DSA) approaches in the IAEA documents is defined to apply as conservative and best-estimate methods in four separate options. Option 4 includes the approach of using best-estimate codes with realistic input data for safety-related systems. Therefore, this approach is interpreted as the realistic approach or Risk-Informed approach by the IAEA, which is derived from the assumptions of the Probabilistic Safety Assessment (PSA). The IAEA documents mean that the PSA is quite the best estimate. However, the basic question that arises is whether the current PSA is the best estimate. Therefore, it is needed to carefully generalize and interpret the best-estimated meaning of PSA. That is to say, although the PSA structure and configuration are similar to the real plant layout, conservative assumptions can be more than the DSA. In other words, the PSA approach historically was introduced to eliminate existing deficiencies in the conservative DSA from the 1980s to 2000s, but the progress of DSA methods has been better in recent years. As an example, the BEPU approach as a matured method officially has been applied in the nuclear power plant licensing process. The scope and level of detail are among the most important shortcomings of the PSA. Although the demonstration of conventional PSA in the framework of the event trees seems realistic and in accordance with the details of the plant, the uncertainty sources in PSA constituting elements like an unclear definition of the initiating event, lack of details about failure data, rough prediction of the operator behaviour and the conservatism in the success criteria definition cause to go away from reality. Besides, the validation procedures for those PSA elements are questionable. It is difficult to justify the conservative data level for the quantification of the PSA output metrics. Thus, the existence of extremely conservative data has caused the PSA cannot to be a panacea for nuclear safety technology, and the individual application of PSA is not a judicious manner. Since the beginning of 2000 up to days, NRC and IAEA emphasize to make decisions in safety-related activities by considering the risk-Informed approach (probabilistic-deterministic fashion). This paper identifies the conservative assumptions in the PSA development and proposes a solution to find a way to map the BEPU approach in the conventional PSA method by defining the validation and uncertainty quantification steps. The result is a flowchart with the identification of cross-links among elements and key features of metrics needed for PSA validation.

707

# Experimental Measurements of Thermophysical Properties of Several Corium Compositions and Influence on Fuel-Coolant Interaction

Caroline Denier[1], Emmanuel de Bilbao[2], Jules Delacroix[1], Pascal Piluso[1]

[1] CEA, DES, IRESNE, DTN, SMTA, Severe Accident Experimental Laboratory, France

[2] CNRS, CEMHTI UPR3079, Univ. Orléans, France

caroline.denier@cea.fr

During a nuclear severe accident, the melting of the fuel leads to the formation of a complex mixture, the so-called corium. Corium thermophysical properties are of major importance for all severe accident phenomena and their knowledge is still today partial. To reduce uncertainties in severe accident codes simulation, a more reliable knowledge of corium properties is necessary whereas very few data are existing. For example for surface tension, some data are available for pure zirconium and very few for the main corium mixture U1-x, ZrxO2-y.

In order to improve knowledge on corium thermophysical properties, a specific experimental device VITI-MBP- (CEA-IRESNE-PLINIUS platform) has been developed. The PLINIUS severe accident platform, located at CEA Cadarache, is dedicated to experimental studies concerning corium and its interactions. It involves several specifically designed facilities, able to reach high temperatures. In particular, the experimental device so-called VITI is dedicated to study thermophysical properties of materials.

In this work, the experimental approach is based on the Maximum Bubble Pressure (MBP) method to measure the surface tension and the density at very high temperature (T>2000 K). Several in-vessel corium compositions have been considered for this study with different rate of oxidation: from corium C-0 –no oxidation of zirconium- till corium C-100 –all zirconium has been oxidized. Results on density and surface tension of five in-vessel corium compositions and their assessment are presented. In order to illustrate the impact of corium thermophysical properties on one severe accident phenomenon, Fuel-Coolant Interaction (FCI) has been chosen using new data obtained on surface tension and density.

708

## Modelling approach for premixing phase in combination of melt jet breakup and premixed layer formation of melt spread

Janez Kokalj, Mitja Uršič, Matjaž Leskovar

Institut "Jožef Stefan", Slovenia

janez.kokalj@ijs.si

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.

An intertwined melt-jet coolant pool and stratified configuration is a realistic condition. However, past research was devoted to either melt-jet coolant pool configuration or stratified configuration and thus an important uncertainty regarding vapour explosion assessment raised.

First objective is to analyse the vapour explosion experiments in combined stratified and melt jet configurations to improve the understanding of fuel-coolant interaction. Secondly, modelling of melt-coolant mixing prior to vapour explosions, which largely defines the amount of melt, participating in the vapour explosions, is being studied.

The developed modelling approach, based on the evaluation of the models for the individual phenomenon enables the estimation of the premixing phase in combination of melt jet breakup and premixed layer formation of melt spread, which is of high importance in nuclear safety.

709

## Organic chemistry of tellurium in severe accident scenarios

Anna-Elina Pasi

VTT Technical Research Centre of Finland, Finland

anna-elina.pasi@vtt.fi

One of the understudied radionuclides released in severe nuclear reactor accident scenarios is tellurium. The released activities of the main tellurium isotope, 132Te, from the two major accidents, Chernobyl and Fukushima, are of the same magnitude as 131I which can be considered the most significant radionuclide released. Moreover, although several tellurium isotopes decay to iodine and consequently contribute indirectly to the increased risk of thyroid cancer, tellurium has received a fraction of the attention in severe accident research compared to the other fission products iodine and cesium. The radionuclides released in high activities are those most volatile. The volatility of an element is significantly affected by its physical and chemical characteristics or speciation. One of the recognized concerns for fission product volatilization is the formation of organic fission product species. Organic material can be present in the containment gas and aqueous phases and originate from e.g. painted surfaces, insulation materials or resins. Iodine is known to form several organic iodides (methyl, ethyl, isopropyl iodide), and until recently other organic fission product species had not been recognized. However, novel results have shown evidence for the formation of organic tellurides from paint constituents in the containment sump conditions. Various organic tellurides (diisopropyl, methyl-isopropyl telluride) were found to form under gamma irradiation in simulated containment sump solution containing tellurium and different paint solvents (texanol, methyl isobutyl ketone). In addition to the formation in the sump conditions, increased transport of tellurium has also been observed in the gas phase in the presence of organic material, suggesting interactions leading to the formation of volatile tellurium species. This work presents the novel results for organic tellurium chemistry in severe accident scenarios and highlights the need for future work.

710

# Analysis of analogue data transmission of dose rate measurements over the HAM amateur radio network in the event of an emergency and possible failure of other communication channels

Thomas Breznik, Benjamin Zorko, Miha Mihovilovič

Institut "Jožef Stefan", Slovenia

thomas.breznik@ijs.si

The general use of radioactive materials (Hazard Class 7) is a dynamic process subjected to changing conditions and unpredictable events. Therefore, it represents a major problem worldwide. The released radioactive substances (gases, aerosols, deposits) pose a social and health threat not only to the local population but present also a great challenge to professional rescue and other intervention teams in the measured data transfer. All the above requires a good preparedness state in readiness and report system of the existing notification system of the National Action Plan. The question itself is thus a prelude to the verification of the adequacy of the existing state of the APRS system for the transmission of ARON-ELME data at a national state level. In case of the emergency event arising and probable failure of other digital communication channels for sending and receiving from in situ field via HAM (Amateur Radio Network) we will present the process and final results of the analysis of the current situation of the stationary network parameters and coverage of the existing HAM system. On the basis of the collected data a GAP Analysis of the information system will follow that will examine the strengths and weaknesses of the system and identify the necessary steps to move from the existing status quo to the desired situation. If the existing system, as well as the plan for collection, intervention and response to emergency can be better organized, this offers an opportunity for further improvements of the autonomous off grid internet system for display and storage of spatial oriented data in the field thus enabling:

- simultaneous accessibility to multiple users,
- minimal user SDR interface,
- built-in VHF receiver.
- storage of data in a MySQL database
- local open street maps as Wi-Fi hotspot, etc.

711

# PWROG SAMG Validation at NEK

Robi Jalovec[1], Mirko Bevc[1], Arunas Bieliauskas[2], Amparo Giner[3], Kevin Honath[4], Laura Genutis[4]

[1] Nuklearna elektrarna Krško d.o.o., Slovenija

[2] Westinghouse Electric Belgium S.A., Belgium

[3] Westinghouse Spain, Spain

[4] PWROG, Westinghouse Electric Company, USA

robi.jalovec@nek.si

PWROG developed a new generic SAMG package, which combines generic SAMG package applicable to U.S. PWRs and generic SAMG package applicable for International Plants. The International plant SAMG are designed for use by PWR plants which typically have installed systems (such as passive Severe Accident hydrogen control and containment filtered containment venting) not currently installed in US plants. In addition, they provide features such as guidance for complete loss of d.c. power and extended treatment of severe accidents occurring from shutdown initial conditions. The U.S. SAMG package includes other enhanced features, such as convenient format, supplemental information included inside severe accident guidelines for technical support center and included detailed benefit consequence information in technical support guideline.

When plant specific guidelines are to be based on a generic owner's group or vendor approach, it is a requirement to perform both verification and validation of the generic guidelines before or during the conversion into plant specific SAMG.

The Krško NPP full scope simulator is capable of modelling severe accident conditions and therefore provides a highly effective platform for the testing of these guidelines, since it seamlessly transitions to Severe Accident conditions, and includes modelling of phenomena (core melt progression, hydrogen, vessel failure and containment challenges) which are often not modeled in simulators.

By using the Krško severe accident simulator for PWROG SAMG validation, a wide range of SAMG was covered, including loss of DC power/instrumentation, degraded PAR condition, use of CFVS for overpressure challenge, severe accident during shutdown in parallel with the SFP event, etc.

In 2023 the Krško SAMG were upgraded to new generic PWROG SAMG package prior to the validation and were used to validate the international plant SAMG features added to the new generic PWROG SAMG.

During the validation, feedback from the Krško MRC and TSC and comments from international observers (PWROG, Westinghouse, Ringhals NPP and Barakah NPP) were used to further improve the new revision of the PWROG SAMG.

712

# Advanced Analytical and Numerical Modelling of ATF FeCrAl and Cr-Coated Zr-Based Cladding High Temperature Oxidation in Steam Atmosphere

Alexander Vasiliev

Nuclear Safety Institute (IBRAE), Russian Federation

vasil@ibrae.ac.ru

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.

Several perspective ATF cladding candidates are chosen for possible application in commercial nuclear power plants (NPPs) in the world including a cladding from FeCrAl alloy and zirconium-based cladding with protective chromium coating (Zr/Cr cladding).

The FeCrAl alloy and chromium 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). The formation of melt leads to acceleration of oxidation and hydrogen generation. 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. In particular, a role of Cr-Zr interdiffusion with subsequent influence on degradation of protective properties is revealed.

In this paper, the new advanced models of high-temperature oxidation of FeCrAl and Zr/Cr cladding are developed. In particular, the Zr/Cr cladding oxidation model is based on simultaneous solution of oxygen and zirconium diffusion equations in different layers of the cladding. A very important role of Zr outward diffusion to the interface between chromium oxide and metallic chromium resulting to severe degradation of protective properties is discovered recently. This phenomenon is taken into account in the model. The models are implemented to newly developed severe accident computer running code.

The comparison of calculated results for FeCrAl and Zr/Cr cladding high temperature oxidation 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 and chromium-coated Zr-based cladding may be optimistic for considerable upgrade of safety level for NPPs especially for design-basis-accident conditions.

713

# Uncertainty Analysis of Severe Accident Scenario in Krško NPP

Matjaž Leskovar, Mitja Uršič, Janez Kokalj

Institut "Jožef Stefan", Slovenia

matjaz.leskovar@ijs.si

The uncertainty estimation for assessing the figure-of-merits characterizing the evolution of a severe accident scenario is a topic of current investigation in the development of the best-estimate plus uncertainty methodology (BEPU). The BEPU approach has become an internationally accepted method for assessing safety margins for a range of high-consequence systems. This approach has increasingly been adopted by the international nuclear safety community to characterize the true safety margin and remove analysis conservatism. The probabilistic method to propagate the input uncertainty is one of the methodologies used to develop uncertainty analyses. Using this methodology, uncertainty analyses are performed by sampling probabilistic distributions that describe the range of possible values that computer simulation model input parameters can have. For each sample of a set of uncertain input parameters, a computer simulation is performed. From the range of code simulation results obtained for each input realization, a distribution of code results is obtained. In this process, the distribution of input uncertainties is propagated to obtain the distribution of code results uncertainty.

Applying the MELCOR code version 2.2 an uncertainty analysis of a severe accident scenario in the Krško NPP was performed. As the initiating event, a strong earthquake was considered, resulting in a simultaneous station black-out and large break loss-of-coolant accident. It was assumed that the active safety systems are not operable and the following passive safety systems were considered: accumulators, passive autocatalytic recombiners and passive containment filtered venting system. The chosen figure-of-merits for the uncertainty analysis are the environmental releases of caesium and iodine, the heat loads on the containment filters, and the in-vessel hydrogen production. The uncertain input parameters were selected based on the State-of-the-Art Reactor Consequence Analysis performed recently by US NRC. The following subset of uncertain parameters was considered: zircaloy melt breakout temperature, molten clad drainage rate, effective eutectic melting temperature, chemical form of iodine and chemical form of caesium. In the paper, the results of the performed uncertainty analysis will be presented and discussed.

714

# Fission Products Behaviour with Respect to the Source Term in Severe Accident Conditions

Yves Pontillona

CEA, France

yves.pontillon@cea.fr

In the case of a hypothetical Severe Accident (SA) in a light water reactor - loss of coolant, failure of safety systems, core degradation with fuel meltdown, … - Fission Products (FPs) and other radioactive materials would be swept out into the containment and, in the event of containment failure, radioactivity would be released into the environment. The knowledge of the ratio of radioactivity released from the core to the environment, i.e. the determination of the “source term”, is of prime and crucial importance for the overall assessment of nuclear power plant safety. Partly due to the impact of the TMI (Three Mile Island, 1977), Chernobyl (1986) and Fukushima (2011) accidents, and to better assess the consequences of such situations, a number of R&D programs have been conducted in this area of research throughout the world. In France, the IRSN (Nuclear Radioprotection and Safety Institute) and EdF (Electricité de France) in collaboration with the CEA (Commissariat à l’Energie Atomique et aux Energies Alternatives) have initiated several experimental programs devoted to the source term of fission products and actinides released from PWR fuel samples in SA conditions.

Within this framework, technical facilities, set up in shielded hot cells, have been developed around the so-called “VERCORS/VERDON” programs. Theirs aims were to quantify the release kinetics of FPs from irradiated nuclear ceramics, quantify the nature of the vapors and aerosols emitted (particle size analysis and chemical composition), and understand how the fuel degrades. The corresponding analytical experiments, conducted in a shielded hot cell, can be considered as complementary to in-pile integral experimental programs such as “PHEBUS FP” and similar to analytical ones conducted in other countries (i.e. the HI/VI program in the USA, the CRL program in Canada and the VEGA program in Japan).

In this context, this presentation gives an overview of the main results gained thanks to the VERDON and VERCORS programs, with a special emphasis on the effect of the oxygen potential on fuel behavior and FPs speciation, with a special focus on the Cs, Mo and Ba chemical species. We will detail how an original and recently developed methodology - based on PWR irradiated and specifically synthetized SIMFUELs samples, together with accurate pre and posttest FP characterizations performed by classical (MEB, EPMA, SIMS…) and XAS examinations – allows FPs speciation to be investigated and its impact on the release to be evaluated. Corresponding experimental data are provided in reducing and oxidizing atmospheres at different stages of a severe accidental sequence. They are intended to be used for assessing the accuracy of the models used in calculation codes such as the fuel performance code ALCYONE co-developed by CEA-EDF-FRAMATOME within the PLEIADES computational environment. Originally devoted to nominal irradiation conditions only, it has been very recently adapted first to accidental conditions (LOCA, RIA), then to SA conditions in order to investigate the mechanisms involved in FPs release. Finally, the well-known FPs classification into four main volatility classes will then be detailed and its usefulness, in assessing Fukushima’s Daiichi accident progression, presented and discussed.

# *Nuclear Education, Training, Workforce Planning, Leadership and Talent Development*

801

## Impact of the ENEN2plus project after one year of implementation

Gabriel Lazaro Pavel[1], Karen Daifuku[2], Michèle Coeck[3], Csilla Pesznyak[4], Christian Schonfelder, Leon Cizelj[5]

[1] European Nuclear Education Network, Belgium

[2] l'Institut international de l'énergie nucléaire, France

[3] Belgian Nuclear Research Center SCK CEN, Belgium

[4] Budapesti Műszaki és Gazdaságtudományi Egyetem, Hungary

[5] Institut "Jožef Stefan", Slovenia

gabriel.pavel@enen.eu

Pressurized thermal shock (PTS) analyses are required to assure that, during the operation of a pressurized water reactor, potentially existing crack-like flaws in the reactor pressure vessel (RPV) wall will not initiate and propagate during loss-of-coolant-accidents (LOCAs) or other PTS-relevant transient scenarios. A finite-element model (FEM) of the RPV is employed in a thermo-mechanical analysis to evaluate the temperatures and stresses in the RPV wall during the transient. For models that include a postulated crack, stress intensity factors (SIFs) are obtained together with other results in a fracture-mechanics analysis. The inconvenience of the latter is that different models of the entire RPV are required to study different crack shapes, sizes or orientations. Additionally to the fact that meshing of the crack is a rather complex task in itself, consideration of the relatively small crack typically also complicates the meshes and increases the overall number of elements in the entire RPV model. To avoid this, the submodeling technique is used, where the thermo-mechanical analysis is performed with the FEM model of the entire RPV without cracks, and submodels of a small portion of the RPV containing the cracks are used in separate fracture-mechanical analyses. The displacements obtained in the thermo-mechanical analysis at the submodel boundary surfaces are employed as boundary conditions.

802

## Capture, transfer and dissemination of knowledge at NEK

Matjaž Žvar

Nuklearna elektrarna Krško d.o.o., Slovenia

matjaz.zvar@nek.si

Knowledge management (KM) is a process through which organizations generate value from their intellectual and knowledge-based assets. In nuclear, it is a crucial aspect of ensuring safety and efficiency, therefor it aims to address the challenges of knowledge loss due to retirement or attrition of experienced staff, knowledge gaps due to lack of adequate training or documentation and knowledge fragmentation due to organizational barriers. Unless Knowledge Management is integrated into work processes, it may not be effectively implemented.

At NEK different approaches and already established processes are used to integrate the knowledge management with the aim on introducing as low as reasonably achievable burden on employees. Systematic approach to training (SAT), knowledge matrices, information technologies, shadowing, …, are just a few, but powerful tools to retain and disseminate knowledge among the employees. Some of them are innovative in their simplicity.

The paper describes the main processes at NEK to capture, transfer and disseminate the knowledge and the results obtained by the KM process throughout the 40 years of operation.

803

## Nuclear research activities from the front to the back end at Centre for Energy Research, Budapest, Hungary

Barbara Somfai, Zoltán Hózer, Margit Fábián

Nuclear research activities from the front to the back end at Centre for Energy Research, Hungary

somfai.barbara@ek-cer.hu

There are four nuclear power plant units (each 500 MW) operating at Paks NPP, Hungary and the construction license with conditions issued for two VVER-1200 reactors by HAEA in 2022. In the near future the new SMR designs and possible implementation will be investigated. The Centre for Energy Research supports the NPP with experiments and modelling to introduce new technologies.

Our research helped to introduce a new cladding with a smaller wall thickness optimized for the water-uranium ratio. Various mechanical tests were carried out to investigate the new cladding properties in different pre-treated (e.g. oxidized, hydrogenated) states. For the mechanical tests the samples are oxidized in different ways and then burst, Mandrel, creep, tensile, four-point bending or ring compression test are performed on them. The microstructural changes can be followed with optical microscope (LOM) or scanning electron microscope (SEM).

Separate effect tests were done with different types of claddings (commercial and accident tolerant) and different coatings. Integral tests were performed with electrically heated fuel bundles for simulation of reactor incidents and accidents up to 2000 °C. During the experiments on-line measurement is done to be able to follow the required parameters.

Different codes (FRAPTRAN, TRANSURANUS, FUROM) are applied in safety analyses and post-test analyses of the experiments with VVER specific model developments.

Research activities on the back end are also performed. The main reactive mechanisms of glass-metal (carbon steel/copper)- bentonite/clay/concrete interfaces are carried out on the experimentally characterized in different environments, focusing on the Boda Clayston Formation disposal concept (glass as waste immobilization matrix, bentonite/cement-based materials as buffer or construction material, clay based materials as buffer/host rocks, metallic materials as container material). Characterization of the chemical evolution in the waste package form. Experimental studies on a mock-up, using a glassy conditioning matrix-studied in different physical and chemical environments.

---

804

## Towards Optimized Use of Research Reactors in Europe - Summary and Results

Joerg Starflinger[1], Markus Hofer[1], Roberta Cirillo[2], Gabriel Lazaro Pavel[2]

[1] University of Stuttgart, Institute of Nuclear Techology an Energy Systems (IKE), Germany

[2] European Nuclear Education Network (ENEN), Belgium

joerg.starflinger@ike.uni-stuttgart.de

The TOURR project is a response to the challenge of coordinating the optimization of the exploitation of available research reactors in Europe. Therefore, its primary objective is to develop an overall strategy for research reactors in Europe and prepare the ground for its implementation.

This strategy is linked with an assessment of the current status of the European research reactors fleet and an estimation of future needs. This includes a plan for a possible upgrade of the research reactor fleet and a plan to maintain the fleet. Based upon the results tools for optimal use of the research reactors fleet will develop and awareness of decision-makers and the public on the role of research reactors will be risen.

In order to build up a strategy, the degree of exploitation in different RR applications was determined using a questionnaire. Initial data acquired from the questionnaire is available in D1.1 – "Database of the European RR fleet" on the ENEN Website. Gaps and opportunities have been identified via statistical analysis and interpretation of the data.

A SWOT (Strengths, Weaknesses, Opportunities, and Threats) revealed that the RR in Europe is diverse and this is a richness. It also implies though that different facilities have different priorities and applications. Improving exchange and communication across the fleet is essential. Furthermore, an overall lack of personnel had been pointed out by the RR which answered the questionnaire. Education and Training in nuclear remain of utmost importance to prevent losing the know-how and keep the nuclear research sector alive and competitive.

The paper/poster summarized the main findings of the TOURR project obtained so far and indicate future steps.

805

## How to attract new talents in nuclear industry

Yannick Le Gonidec

APAVE, France

yannick.legonidec@apave.com

Nuclear projects are blooming in the world. Due to these opportunities ,the nuclear supply chain challenges are now to attract new talents in nuclear industry.

Our contribution includes training materials for students, workers outside of nuclear industry, nuclear supply chain job training in order to give them a better understanding of the nuclear industry and focus on safety.

Our training demo will be summarize though 3 serious games. These serious games could be use by the teacher at school with students and or for the nuclear suppliers to introduce nuclear industry. Further this demo our knowledge management platform can be use to develop specific on job training.

Our training materials are designed to prepare and improve safety during work execution on site whether it be for new build project or operating NPP.

806

## Basics of Nuclear Technology Courses in Nuclear Training Centre Ljubljana

Tomaž Skobe

Institut "Jožef Stefan", Slovenia

tomaz.skobe@ijs.si

The paper presents experiences from performing nuclear technology courses at 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 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.

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 2023 the 45th edition of the course was conducted.
The paper will present the Basics of Nuclear Technology course organization, materials preparation, course content and results, and feedback from participants.

807

## When Is The Best Time for Configuration Management in Nuclear Industry?

Barcza István

Barcza István, Hexagon - Asset LifeCycle Intelligence, Hungary

istvan.barcza@hexagon.com

Configuration Management is a set of processes, practices, and tools necessary to effectively manage all the obligations regarding the design, construction, operation, maintenance, and decommissioning of a NNPP, thereby maintaining consistency between the design and licensing basis requirements (what needs to be there), the physical configuration (what actually is there), and the configuration information (what we say is there).

Design & Licensing Basis Information (DLBI) is one of the cornerstones of configuration management. Owners and operators also have an economic incentive to develop an approach that will reduce the plant staff effort required to manage the DLBI and to ensure that the plant is not vulnerable to lost revenues resulting from regulatory concerns over the integrity of the DLBI and loss of controlling change.

808

## Knowledge Management activities in the EURAD programme

Nadja Železnik

Elektroinštitut Milan Vidmar, Slovenia

nadja.zeleznik@eimv.si

Particular emphasis is dedicated to Knowledge Management (KM) activities within the EURAD (European Joint Programme on Radioactive Waste Management) programme to ensure the capture of existing knowledge, transfer of knowledge between Members States and management of the knowledge for future generations. Within EURAD there are a variety of tools and methods that support knowledge management activities with dedicated work packages (WP): WP1 EURAD roadmap – activities to orientate people 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; WP11 State of Knowledge - 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 - Activities consisting of developing a comprehensive suite of instructional guidance documents that can be used by Member-States with RWM programmes and WP13 Training and Mobility - Activities consisting 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 and plans for future work.

809

## Youngsters about Nuclear Energy – Year 2023 Poll

Radko Istenič, Igor Jenčič

Institut "Jožef Stefan", Slovenia

radko.istenic@ijs.si

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 8000 visitors per year. After limitations imposed by covid-19 pandemic in 2020 and 2021 the number decreased but recovered in 2022 and 2023. 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.

---

810

## Public Opinion about Nuclear Energy – A Comparison between Youngsters and General Population, Year 2023 Poll

Radko Istenič, Igor Jenčič

Institut "Jožef Stefan", Slovenia

radko.istenic@ijs.si

One of the important activities of the Nuclear Training Centre at the Jožef Stefan Institute is public information about energy, electricity, nuclear power and nuclear technology, radioactivity, and fusion.
Our main target population are the schoolchildren from the last grades of elementary school and from high school (ages 13-18). Since 1993 we monitor their knowledge and opinion by polling some 1000 youngsters each year. These polls were intended as a guidance to our information activities, as well as to detect trends in their opinion and to a lesser extent as a representation of public opinion about nuclear energy in general. The polls were always conducted before the lecture or visit to the exhibition of nuclear energy, to get their unbiased opinion. We found that their basic knowledge of energy issues and their attitude towards nuclear energy, which is relatively favourable, does not change much over the years. Also, the share of undecided is always relatively high.

This year we decided to poll for the first time the general population with the same set of 10 questions. The poll was conducted by a professional polling agency on a representative sample of general population in Slovenia (1000 polled).

The results of the poll show that among the general population the support for nuclear energy and for a new NPP in Slovenia is much higher than among youngsters. The shares of those opposed and undecided are lower than in the group of our visitors.

811

## Renovation of the permanent exhibition on nuclear technology

Vesna Slapar Borišek, Igor Jenčič, Tomaž Skobe

Institut "Jožef Stefan", Slovenia

vesna.slapar-borisek@ijs.si

An important activity of the Nuclear Training Centre ICJT is public information about energy and nuclear technology. It is composed of a short lecture, a demonstration of radioactivity experiments, and a visit to the permanent exhibition on nuclear technology. The first exhibition was set up in 1992, and around the year 2000, it was expanded and graphically redesigned. After two decades, we felt that it was time to improve the concept of the exhibition by extending the content and by including comprehensive information on all kinds of energy sources and their impacts on electricity production and the environment.

The new exhibition was prepared in collaboration with professional designers, to make it more modern and attractive for the younger population, which are the majority of our visitors. In this context, the content is presented in an interactive manner as much as possible.

From February to May 2023, the new exhibition was visited by 4200 youngsters with their teachers and their response is very positive.

812

## Vibration impact in nuclear power generation: Go-viking advancements

Kateryna Piliuhina[1], Angel Papukchiev[2], Kevin Zwijsen[3], Philippe Planquart[4]

[1] European Nuclear Education Network, Belgium

[2] GRS gGmbH – Gesellschaft fur Anlagen und Reaktorsicherheit, Germany

[3] NRG – Nuclear research and consultancy group, Netherlands

[4] VKI – Von Karman Institute for Fluid Dynamics, Belgium

vesna.slapar-borisek@ijs.si

An important activity of the Nuclear Training Centre ICJT is public information about energy and nuclear technology. It is composed of a short lecture, a demonstration of radioactivity experiments, and a visit to the permanent exhibition on nuclear technology. The first exhibition was set up in 1992, and around the year 2000, it was expanded and graphically redesigned. After two decades, we felt that it was time to improve the concept of the exhibition by extending the content and by including comprehensive information on all kinds of energy sources and their impacts on electricity production and the environment.

# *Reactor Physics and Research Reactors*

901

# Update on the review of more than 15 years of JSI – CEA fruitful collaboration on nuclear instrumentation developments

Christophe Destouches[1], Luka Snoj[2], Loic Barbot[1], Vladimir Radulović[2], Nicolas Thiollay[1]

[1] CEA-IRESNE-DER, France

[2] Institut "Jožef Stefan", Slovenia

christophe.destouches@cea.fr

The collaboration on nuclear instrumentation between the Reactor Physics Division of the Jožef Stephan Institute (JSI) and the CEA Experimental Physics Section (Experimental physics, Safety experiment and Instrumentation Section since January 2018) started in 2008 in the framework of multiple bilateral agreements between the CEA and the Slovenian Ministry of Higher Education, Science and Technology.

This collaboration focused through successive biannual projects on Miniature Fission Chambers, Self-Powered Neutron Detector and Gamma (Ionization Chambers and Thermoluminescent Detectors) measurement techniques as well as kinetic parameter measurement techniques (beff), reactor dosimetry unfolding techniques, nuclear data, calorimetry measurements.

These different projects have contributed to test and validation of several CEA measurement devices and systems (MONACO, FNDS, and SPECTRON) and modelling codes (CALMAR, MATiSSe). On JSI side, measurement campaigns performed with this instrumentation allowed improving the JSI unfolding code, performing experimental validation of computational codes and models developed at JSI, and a thorough characterization of the neutron fields in the TRIGA Mark II reactor, including the reactor pulse operating mode. These activities resulted in publishing in 2016 the evaluated fission rate experimental benchmark of the JSI TRIGA reactor in the OECD/NEA International Handbook of Evaluated Reactor Physics Benchmark Experiments.

This collaboration has also increased international visibility of both organisms with the publication of more than 35 common papers in international conferences or in scientific peer reviews including the common participation to several scientific committees (ANIMMA, ISRD, NENE).

This paper presents an update of the review paper presented in NENE 2018 including recent and ongoing projects. A discussion about future projects and the interest of a future Slovenian research reactor will conclude this paper.

902

# Uncertainty Propagation of decay data for Decay Heat Calculations: A Monte Carlo Approach

Yohannes Molla, Lydie Giot, David Laks

Laboratoire de Physique Subatomique et des Technologies associées (Subatech), France

yohannes.molla@subatech.in2p3.fr

Decay heat is the residual power produced by radioactive fission product nuclei, minor actinides and delayed neutron fission after a nuclear reactor is shut down. This heat decreases with time, but it can still release a significant amount of power immediately after shut down. As a result, decay heat must be taken into account during unloading, transportation, reprocessing, and final repository design of spent nuclear fuel. Moreover, it plays an important role for emergency safety system design of Gen IV reactors.

Computationally, decay heat can be determined by the summation method. This method is an extensive approach which involves calculating the contribution of each radioactive nuclide and summing them to obtain the total decay heat. The individual decay heat contribution for each isotope is calculated as the product of its concentration as a function of cooling time, decay constant and mean decay energy, which are nuclear decay data. The decay data are generated from processed and evaluated experimental data. Therefore, these data inherently carry uncertainties that propagate into the decay heat calculation. Quantifying the impact of these uncertainties on the decay heat calculation is crucial for optimizing the design and safety margins of new reactor concepts and spent nuclear fuel disposal casks.

To this end, the aim of this work is to quantify the uncertainty propagation of decay data on decay heat via a Monte Carlo approach. An in-house code named COCODRILO written in Python and coupled to the SERPENT2 depletion code was developed for this purpose. The approach involves sampling from fission yields and decay data using the Gaussian sampling method and/or nuclear data covariance matrices. Moreover, the code is flexible in that it is capable of sampling the mean energies of beta particles and electromagnetic emissions separately if needed and for some dedicated nuclei. First results on fission pulses decay heat cases for U/Pu cycle will be presented.

903

# The ISOLPHARM project: 111Ag Production and separation at L.E.N.A.

Federico Alfinito[1], Luca Morselli[3], Antonietta Donzella[5], Alberto Arzenton[8], Mattia Asti[9], Silva Bortolussi[6], Stefano Corradetti[3], Giancarlo D'Agostino[7], Marco Di Luzio[7], Matteo Ferrari[10], Andrea Gandini[1], Marianna Tosato[2], Valerio Di Marco[2], Marcello Lunardon[4], Valerio Villa[5], Andrea Salvini[1], Lisa Zangrando[4], Aldo Zenoni[5], Alberto Andrighetto[3]

[1] Laboratorio per l'Energia Nucleare Applicata, Italy

[2] UNIPD - Dipartimento di Scienza Chimiche, Italy

[3] INFN, Laboratori Nazionali di Legnaro, Italy

[4] INFN, Sezione di Padova, Italy

[5] Dipartimento di Ingegneria Meccanica e Industriale, Università degli Studi di Brescia, Italy

[6] Dipartimento di Fisica, Università degli Studi di Pavia, Italy

[7] Università degli Studi di Pavia, Istituto Nazionale di Ricerca Metrologica, Dipartimento di Chimica Generale, Italy

[8] Università degli Studi di Siena, Dipartimento di Scienze Fisiche della Terra e dell'Ambiente, Italy

[9] Radiopharmaceutical Chemistry Section, Nuclear Medicine Unit, AUSL-IRCCS di Reggio Emilia, Italy

[10] Université Jean Monnet, Laboratoire Hubert Curien, CNRS, UMR 5516, France

federico.alfinito@unipv.it

The ISOLPHARM project is aimed to produce and test new radionuclides for cancer treatment and diagnosis. The idea is to exploit the isotope separation on-line (ISOL) technique to produce unconventional radionuclides that are difficult to obtain with standard production techniques. Amongst different radionuclides of interest 111Ag is particularly promising for therapy, as a beta emitter with a range of about 1.8mm, and a lifetime of 7.45 days.

Until the SPES cyclotron is completed at the Legnaro laboratories, production of 111Ag is possible through the Neutron Activation process of both natural and enriched 110Pd. Within this context is crucial the experimentation at the Laboratory of Applied Nuclear Energy (L.E.N.A) in which a TRIGA MK II research reactor is in operation since 1965. This process is important for both the activation of 110Pd and the radiochemical isolation of Silver from the target. Radiochemistry experiments focused on the dissolution of the irradiated target where 111Ag is obtained are required to optimize the radiochemical and radionuclidic purity. In Addition, considering the high enrichment costs of 110Pd a recovery protocol for the irradiated samples must be considered during the purification process.

In this work will be presented the network of the ISOLPHARM project, the description of milestones for its subprojects EIRA and ADMIRAL and the data collected at the L.E.N.A laboratory.

904

# FEniCSx-OpenMC Coupling for Neutronic Calculation with Temperature Feedback

Stefano Riva[1], Lorenzo Loi[1], Carolina Introini[1], Antonio Cammi[1], Xiang Wang[2]

[1] Politecnico di Milano, Department of Energy, CeSNEF - Nuclear Engineering Division, Italy

[2] Harbin Engineering University, College of Nuclear Science and Technology, China

stefano.riva@polimi.it

The state of an operating nuclear reactor depends on several interdependent physical phenomena, which can be considered simultaneously by modelling the system using a multi-physics (MP) approach. MP allows a higher level of detail of the system's properties at the expense of code complexity and computational burden, whereas, in the past, single-physics codes were dominant due to the limited computational resources, and the coupling effects were typically introduced using correlations or boundary conditions to the problem. In the context of nuclear reactors, the fundamental coupling is between neutron physics and thermal hydraulics, as their interaction directly affects the power and temperature profiles, which are quantities of interest during both the design and safety analysis phases. Due to the recent improvements in computational capabilities, the MP approach has become feasible, focusing the efforts on the development of interfaces between different single-physics numerical codes (in most cases, already validated and verified) to catch the MP coupling. This work focuses on developing a tool capable of determining the temperature profile of a characteristic fuel pin of a PWR when the power generated by the system is known: this test case is interesting because the thermal feedback effects produce a shift in the power peak compared to the middle of the rod, which a well-known phenomenon in nuclear reactor pins. This tool is developed in a Python environment, using the open-source library FEniCSx for the thermal-hydraulic analysis and the OpenMC Monte Carlo code to describe the fissionable system; this choice has been made to have the whole code inside a single open-source environment which, compared to state-of-the-art proprietary codes, offer higher accessibility and community feedback. In the coupling, an explicit method is applied whose convergence is based on a Picard scheme, using an adaptive relaxation scheme: this strategy is one of the most adopted techniques due to its simplicity; however, monolithic approaches can be also adopted which may give better results even though the implementation phase is more challenging. The proposed coupling approach can predict the peak shift in the power density as per the literature, thus more efforts can be made to extend the current model to more complex cases.

905

## An Innovative Reactivity Control Strategy for Small Modular Reactors

Carolina Introini, Marco Herbas Lopez, Carlo Lombardi, Antonio Cammi

Politecnico di Milano, Department of Energy, CeSNEF - Nuclear Engineering Division, Italy

carolina.introini@polimi.it

Currently, the main reactivity control methods for small modular reactors (SMR) use control rods as the primary control system, with chemical SHIM or burnable absorbers as secondary systems to control the reactivity excess. However, this kind of control system presents some risks, such as control rod undesired drop or positive moderator coefficient due to the high concentration of boric acid in the moderator. This paper evaluates the possibility of using a displaceable heavy neutron reflector with a neutron absorber (boron) as a secondary reactivity control method. The reference NuScale core geometry has been simulated using the DRAGON5 and DONJON5 deterministic codes, computing the neutron flux and power distribution at each reflector withdrawal step. Two different strategies have been considered: 1) withdrawing the entire movable reflector block towards the upper part of the vessel and 2) separating the reflector block into two equal parts, removing each in different directions from the core equator region. Results indicate that the most suitable reflector withdrawal mechanism is the latter: this solution is promising to replace secondary reactivity control methods in small reactor cores.

---

906

## Fast neutron fluence profiling at the JSI TRIGA reactor irradiation facility

Klemen Ambrožič[1], Valentina Sola[2,3], Igor Mandić[1], Marco Ferrero[3], Gregor Kramberger[1], Luka Snoj[1]

[1] Institut "Jožef Stefan", Slovenia

[2] Università degli Studi di Torino, Italy

[3] INFN, Sezione di Torino, Italy

klemen.ambrozic@ijs.si

We present the analysis on the fast neutron fluence profile measurement in the JSI TRIGA mk. II reactor irradiation facility. A number of multi-pad Low-Gain Avalanche Diodes (LGADs) sensors were mounted on a cross-like support structure keeping them at a known position in the irradiation channel. Each of the 7.7 mm × 7.7 mm multi-pad sensors consists of an array of 5 x 5 individual LGADs.

The assembly was inserted into a polyethylene enclosure and irradiated at full power in the F19 irradiation channel to the fluence of 1.53 × 10^15 neq/cm2 where neq stands for 1 MeV equivalent neutron in silicon. The active dopant concertation in LGAD gain layer was extracted via capacitance-voltage measurements before and after irradiation. The active dopant concentration, and therefore the depletion voltage of the gain layer, is reduced by irradiation with fast reactor neutrons due to the so-called acceptor removal effect. The

change of the gain layer depletion voltage was used to precisely measure fluence to which each individual LGAD in the multi-pad sensor was exposed.

To affirm the measured fast neutron fluences by each sensor, the experiment was reproduced by simulations using the Monte Carlo particle transport code MCNP, ENDF/B VII.0 nuclear data libraries and a detailed model of the JSI TRIGA reactor, along with detailed modeling of its configuration (control rod positions, irradiation channels) and the LGAD irradiation assembly. Since we have no information on the assembly's position inside the channel with respect to the assembly axial rotation, two distinct positions were simulated: one, where one of the cross legs points towards the core center (perpendicular), and where cross legs are diagonal. Neutron flux was calculated over the entire chip, as well as at sections corresponding to individual diodes.

Comparison of the computational results with measurements suggest that the perpendicular setup more closely resembles the LGAD assembly’s orientation. Comparison of results computational and experimental results, particularly with respect to the unknown exact position of the LGAD assembly during irradiation shows a reasonable agreement, where computed values are generally within the measurement’s error bars, albeit with some outliers. However, the general trend of the fast neutron flux dispersion is well observed both by measurements and simulations.

We’ve established a fast neutron fluence measuring technique using an array of LGADs and confirmed it by simulations. We showed that high precision mapping of fast neutron flux with millimeter spatial resolution is feasible with relatively simple experimental technique.

---

907

## Influence of the temporal variability of neutron fluxes in the carousel of TRIGA Mark IPR-R1 research reactor, CDTN, Brazil, in the k0-method of neutron activation analysis

Maria Angela de Barros Correia Menezes[1], Rodrigo Reis Moura[1], Radojko Jaćimović[2]

[1] Nuclear Technology Development Center, Brazil

[2] Institut "Jožef Stefan", Slovenia

menezes@cdtn.br

The team of the Laboratory for Neutron Activation Analysis, Nuclear Technology Development Centre (CDTN), Belo Horizonte, Brazil, has been carrying out the k0-standardization method of neutron activation analysis (k0-NAA) for elemental concentration determination since 2003. Diversified samples, like airborne particulate matter, sediment, soil, ore, food, plants and human samples, have been analysed by the method. The samples are usually irradiated for 8 hours in the carousel facility of the 100 kW TRIGA Mark I IPR R1 at CDTN with an average thermal neutron flux of 6.3 × 1011 cm-2 s-1.

The k0-method is a “quasi” absolute technique in which gold monitor replaces the standards of the investigated elements in the sample. Nonetheless, the spectral parameters α (epithermal neutron flux shape factor), f (thermal-to-epithermal neutron flux ratio) and the neutron fluxes are carefully determined.

The method, in its basic structure, considers that the irradiation conditions is continuous and constant. However, during long operation there are flux variations due to several actions to keep the power at 100 kW, for instance, compensation of reactivity change by control rod movement.

The objective of this study was to verify quantitatively the temporal variation of neutron flux in the IC-40 of the carousel facility of the TRIGA reactor. It was determined that the flux variation in the IC-40 is about 15% during 8 hours of irradiation if there is no correction of the control bars. With data from the Au and Zr neutron flux monitors and calculations with the Kayzero for Windows software (KayWin), an equation was built to correct the flux variation in the k0-method. The data obtained in this study for the temporal flux variation and its influence on the results obtained by k0-NAA will be presented and discussed.

This work was partially supported by the Brazilian Foundation for Research Support of Minas Gerais, FAPEMIG, by the Brazilian National Council for Scientific and Technological Development, CNPq, and also had a financial support by the Slovenian Research Agency (ARRS) through programme P1-0143.

---

908

## Influencing Factors of Oxygen Control Performance of Pt/air Electrochemical Oxygen Pump in Liquid LBE

Danna Zhou

International Academy of Neutron Science, China

Shandong Key Laboratory of Neutron Science and Technology, China

danna.zhou@fds.org.cn

The compatibility of Lead-cooled Fast Reactor (LFR) structural material with liquid Lead-Bismuth Eutectic (LBE) can be improved by controlling the dissolved oxygen concentration in the LBE. In this manuscript, Pt/air electrochemical Oxygen Pump (EOP) was used to control the dissolved oxygen concentration in liquid LBE to protect structural materials against corrosion by LBE. The EOP was fabricated using Yttria Partially Stabilized Zirconia (YPSZ) as a solid electrolyte and Pt/air as a reference electrode. The performance of oxygen added/removed of the EOP with different electrode area, applied voltage and LBE temperature was tested in LBEby chronoamperometry.

The experimental results show that the oxygen control efficiency of the EOP increases proportionally with the area of the Pt electrode, LBE temperature and applied voltage. When the oxygen concentration is high, the oxygen removal efficiency of the oxygen pump is mainly limited by the transport speed of oxygen ions in the YSZ solid electrolyte, while when the oxygen concentration is low, it is mainly limited by the mass transfer of oxygen from the LBE to the surface of the YSZ. In addition, the polarization effect will become larger with the increase of the oxygen pump current, so that the oxygen control efficiency of the oxygen pump will decrease.

909

## Possibility of using so-called “binary” “grey” control rods for power regulation of a nuclear reactor

Alejandro Marro

Institut "Jožef Stefan", Slovenia

am17548@student.uni-lj.si

Even though the great majority of nuclear reactors operate at a constant power during most of the time, in the so-called base-load operation mode, the reactor power may undergo changes due to grid frequency requirements or if the reactor operates in a load-following mode. This affects the reactor reactivity through different feedback mechanisms. Among the most common solutions to compensate for that are the use of boric acid or the use of control rods. This last one can cause local perturbations of the neutron flux since its insertion is not homogeneous in the whole core, leading to undesirable phenomena such as time-space oscillations of 135Xe concentration and the modification of the fuel burnup pattern. In this paper, the use of so-called “binary” “grey” rods is proposed as an alternative. Grey rods, which contain materials with lower absorption cross-sections, have a lower reactivity worth than regular, shutdown “black” control rods and are intended to be inserted over the entire height. The idea is to design and study the feasibility of grey control rods with reactivity worths of multiples of 2, i.e. to create a set of rods of binary reactivity worths, and study their feasibility. In this case, the continuous-energy Monte Carlo neutron transport code SERPENT is used to create a 2D model representing a typical PWR reactor core. The temperature defect is calculated from HZP to HFP to define the reactivity worth requirements that need to be fulfilled by the binary control rods. Binary control rods are designed as well as their geometrical configuration in the core. In addition, different material compositions are considered to evaluate its possible effect on reactor criticality. Apart from that, the radial power distribution is studied to evaluate possible perturbations of the flux on a local scale. In this regard, attempts are made to minimize the interference effect between the control rods clusters.

910

## Intermediate range detectors for control rod worth measurements with rod insertion method

Tanja Goričanec, Luka Snoj, Marjan Kromar

Institut "Jožef Stefan", Slovenia

tanja.goricanec@ijs.si

For a safe operation of a nuclear power plant it is important to accurately control the reactivity. Reactivity in a typical pressurized water reactor (i.e. Krško nuclear power plant) is controlled by boric acid dissolved in the water, the control rods, and burnable absorbers in the reactor core. The control rod reactivity worth is a safety related physical parameter and can be determined by calculations and by measurement. It can be measured by different methods, e.g. the boron dilution method, rod swap method or the rod insertion method. The rod insertion method was developed at the Reactor physics department at the Jožef Stefan

Institute and was world novelty at the time of publication. It is based on the analysis of the signal recorded with the ex-core neutron detectors during the continuous insertion of a control rod bank. The control rod reactivity worth is measured for each new core configuration before the start-up. For the rod insertion method power range detectors are currently utilized at the Krško nuclear power plant. Power range detectors cover almost the entire active core height and enable axial averaging of the signal. Their downside is relatively low neutron signal at low reactor powers compared to the background. On contrary, to the power range detectors, the intermediate range detectors utilize compensated ionization chambers, where signal due to the gamma rays can be compensated. Their upside is that their signal is well above the background signal or noise. As the control rod moves, the neutron flux profile in the reactor core also changes. These redistributions can change the reading of the ex-core detectors, resulting in a non-linear power reading and influence control rod reactivity worth determination. To account for radial and axial redistributions, redistribution factors are introduced. Due to the smaller active height intermediate range neutron detectors are more sensitive to the neutron flux redistributions caused by the control rod movement compared to the power range detectors. With accurate determination of neutron flux redistribution factors, the possibility to use intermediate range detectors instead of the power range detectors appears attractive and is analysed within this work. In aim to predict ex-core neutron detector response, a detailed geometrical model of the core and ex-core structures, such as reactor pressure vessel, surrounding concrete and explicitly modelled ex-core neutron detectors were developed to be used with the state-of-the-art Monte Carlo code for neutron transport MCNP. The developed model and methods were used to predict ex-core neutron detector response and its perturbation with the control rod movement. The axial dependant reaction rate redistribution factors were determined and used to correct the measured signal to obtain the control rod worth. Control rod worth measured with the rod insertion method was compared to the boron dilution measurements. A good agreement within 2 % was observed. The use of intermediate range detectors for rod insertion measurements using newly calculated axial dependent reaction rate redistribution factors represent the way to update and improve the rod insertion measurements. This would improve the axial calibration of their reactivity worth and consequently improve the safety of the nuclear power plant.

911

## Characterization of Cherenkov Radiation for Nuclear Power Measurements: A Study at the JSI TRIGA Research Reactor

Julijan Peric[1,2], Vladimir Radulović[1,2], Luka Snoj[1,2]

[1] Institut "Jožef Stefan", Slovenia

[2] University of Ljubljana, Faculty of mathematics and physics, Slovenia

julijan.peric@ijs.si

The Cherenkov power meter is an independent, reliable and cost-effective measurement system based on Cherenkov radiation detection at the Jožef Stefan Institute TRIGA research reactor. The reactor is capable of producing a maximum neutron flux of about 2×10$^{13}$ $n\ cm{-1}$at a thermal power of 250 kW. Reactor power is monitored by five measurement channels equipped with various conventional neutron detectors.

Cherenkov radiation is present in most water-cooled nuclear reactors, with the exception of zero-power reactors. It is caused by energetic charged particles traveling faster than the speed of light in a dielectric

medium. In open-pool reactors, Cherenkov radiation can be observed as a blue glow around the reactor core. The Cherenkov power meter was developed, implemented, and tested at the Jožef Stefan Institute research reactor TRIGA to explore an alternative approach for measuring reactor power. The intensity of Cherenkov light produced in the water due to the prompt gamma rays is in principle proportional to the neutron flux during a reactor operation. By quantifying the intensity of Cherenkov light, it becomes feasible to monitor the neutron flux throughout the operation of the reactor. The developed Cherenkov power meter is based on a closed tube to avoid interference from external light sources and radiation damage to optical fibers. The tube is placed in the periphery of the reactor core, the lower part of the tube is filled with water and serves as a source of Cherenkov light. The intensity of the light reflected from the source on the inner surface of the tube is measured from the top of the tube.

In this paper, the developed Cherenkov power meter is presented as an independent, reliable, and cost-effective measurement system that has great potential for further development and eventual use in future nuclear instrumentation. During the testing process, several promising updates and improvements were identified for the Cherenkov power meter that further strengthen its potential as a measurement system. In particular, advances were noted in the accuracy, sensitivity, and overall performance of the system, indicating opportunities for optimization.

912

## OpenMC Analysis of TRIGA Mark II Reactor Void and Temperature Reactivity Coefficients

Lorenzo Loi, Stefano Riva, Carolina Introini, Antonio Cammi, Enrico Padovani

Politecnico di Milano, Italy

lorenzo.loi@polimi.it

Due to stringent safety requirements for nuclear facilities, there is a need for the development of precise and accurate computational tools for the reactor safety analysis both during the licensing process and during standard operation. Achieving this goal requires verification by different state-of-the-art neutronic codes and validation by comparing simulations with experimental data. The recent development of open-source platforms has increased the interest in adopting such technologies, which, compared to proprietary software, offer continuous exchange between developers and users and direct access to the source code. In safety analyses, studying feedback coefficients is crucial for evaluating the reactor dynamic response to control procedures as well as during accidental scenarios. This kind of analysis provides an in-depth understanding of reactor behavior under different operating conditions (i.e., different power levels) and experimental settings. Existing codes are suited for commercial reactors and may not offer the capabilities for simulating future generation systems. In this framework, this paper analyses the thermal feedback coefficients and void coefficient of the TRIGA Mark II reactor using a Monte Carlo model developed with the Python-based open-source OpenMC code. The reason behind the choice of this particular reactor is twofold: first, this reactor represents a landmark in nuclear research due to its unique asymmetrical configuration and the presence of UZrH fuel, and especially for its passive safety feature, made possible by its highly negative feedback coefficients. Furthermore, a large number of experimental data are readily available for these reactors. This work considers two different scenarios for the validation: the first case is the insertion of positive reactivity through a control rod extraction, allowing the temperature to increase along with the power; the second

case simulates the reactivity perturbation coming from the presence of a void volume (e.g., in sub-cooled boiling regime) through the placement of a sample made of aluminum and filled with air or water in the central channel. The experimental scenarios related to the evaluation of the feedback coefficients are accurately reproduced: tracking the change in the criticality level (k-eigenvalue) versus some physical quantities (i.e. the temperature or the void level) allows for the calculation of the feedback coefficients, and the results obtained from the OpenMC simulation are compared to both the experimental data as well as results from the SERPENT Monte Carlo code, showing a good agreement.

913

## Experimental and Computational Validation of Novel Depletion Algorithm in the RAPID Code System using JSI TRIGA reactor

Anže Pungerčič[1,2], Alireza Haghighat[3], Luka Snoj[1,2]

[1] Institut "Jožef Stefan", Slovenia

[2] University of Ljubljana, Faculty of mathematics and physics, Slovenia

[3] Nuclear Engineering Program, Virginia Tech, USA

anze.pungercic@ijs.si

The accurate determination of physical parameters of a research reactor core is crucial for its utilization, use, and reliable operation. As these parameters change with fuel depletion, it is important to predict their expected changes using reactor calculations. Modern Monte Carlo depletion codes, which offer high accuracy, require intensive computational resources, particularly for 3-D depletion calculations. To address this problem the Virginia Tech Transport Theory Group (VT3G) and the "Jožef Stefan" Institute (JSI) have been in collaboration to develop and validate hybrid algorithms in the RAPID code system, which offer the speed of simplified deterministic calculations with the accuracy of Monte Carlo calculations. This is achieved by using the Multi-stage Response-function particle Transport (MRT) methodology, which divides the problem into multiple stages, consisting of pre-calculating response function/coefficients that describe physical phenomena and the solving the neutron transport equation in real-time using the Fission Matrix (FM) methodology. The bRAPID algorithm for depletion calculation in the RAPID code system uses the same approach by pre-calculating FM coefficients and nuclide composition as a function of fuel burnup and performs on-the-fly linear interpolation between the FM coefficients to obtain 3D pin-power profiles.

The RAPID code system was applied to the JSI TRIGA Mark II research reactor and its steady-state calculations were validated on criticality benchmark core, neutron flux profile measurement using neutron dosimetry and control rod calibration curves. Recently the bRAPID burnup algorithm was applied to JSI TRIGA to validate the burnup simulations, where good agreement with the Serpent-2 code was observed (as average discrepancy of about 37 pcm). Furthermore, capability of accurately calculating 3D fission neutron source distribution changes with burnup was demonstrated by achieving less than 1 % relative differences compared to the Serpent-2 predictions. It was demonstrated that bRAPID accurately determines burnup in areas with high gradients of neutron flux (e.g., vicinity of control rods), which is crucial for small and compact research reactor cores. Burnup simulations were further validated experimentally by analyzing excess reactivity measurements throughout the reactor operational history. The analysis was conducted for three different mixed TRIGA cores, which included all four types of fuel elements, differing in enrichment (LEU – 19.9 % and

HEU – 70 %) and type of cladding (stainless steel and aluminum). Linear reactivity burnup reduction coefficient was determined, and great agreement was observed between Serpent-2 and the RAPID code system, within 1 sigma of measurement uncertainty. Next step in validation was to perform in-direct measurements of individual fuel element burnup using fuel reactivity worth method. In two-day measurement campaign reactivity worth of seven fuel elements at the JSI TRIGA were measured. Fuel elements from different rings and irradiation times were chosen for validation. Burnup calculations and experimental measurement were simulated using Monte Carlo Serpent-2, hybrid RAPID. Excellent agreement in calculated and measured reactivity worth was observed for Serpent-2 and RAPID code.

The paper will present further information regarding the bRAPID methodology, the RAPID code system, and will focus on the experimental comparison briefly described in this abstract. In conclusion the RAPID code system with its bRAPID methodology offers a time efficient and accurate way to calculate 3D pin-wise burnup, nuclide inventory, and power distribution and its changes with fuel burnup. The presented methodology is suited especially for research reactors due to its diverse operations in time, power, and control rod positions.

914

# Burnup-dependent group constant parametrization by applying different machine learning methodologies

Dániel Sebestény, István Panka, Bálint Batki

Centre for Energy Research, Hungary

sebesteny.daniel@ek-cer.hu

In neutronics calculations, full-core Monte Carlo calculations are being increasingly used due to the continuous increase in computing power, however, the traditional two-step calculation schemes are still considered the usual practice in computationally demanding tasks like transient full-core calculations with feedbacks. In these schemes, the first code produces few-group constants spatially homogenized to assemblies or nodes, which the second code uses when calculating the full core. The parametrization of these group constants at the assembly level is an important step of these schemes. While theoretically, the full-core level code could call the assembly-level one each time it requires homogenized few-group constants, it would be time-consuming, so the usual practice is to prepare a parametrized group constant library for each node of the simulated system. It is basically a model that predicts a particular group constant for a specific set of operating conditions (e.g., temperatures, moderator density, burnup, boron concentration). This model requires training on some base points generated by the lower-level code.

While for a simple problem, the easiest solution is to generate as many base points as needed to get the required accuracy using simple (linear) interpolation, it might be beneficial to use other methods when we use a Monte Carlo program as the lower-level code to generate base points, as these codes require much more computational time while also bringing about statistical errors. These two factors can bias the aforementioned simple interpolation approach. Thus, the goal of this work was to develop models that are resistant to statistical errors as well as perform reasonably well on smaller training datasets.

This paper explores the feasibility and advantages of using different types of models ranging from polynomial regression to more advanced machine learning methods like neural networks for group constant parametrization. A Python module was developed that undertakes hyperparameter optimization and automatic model selection. Models were trained and evaluated on group constants of VVER-440 and VVER-1200 fuel assemblies calculated using the Serpent Monte Carlo and MULTICELL 2D transport codes. It was found that the best-performing model is a type of polynomial interpolation, where the maximum degree and the root of each feature as well as the overall maximum degree of any polynomial are the model's hyperparameters. This approach needs a considerably lengthy hyperparameter optimization; however, it can be fine-tuned to the problem and avoid overfitting with the help of regularization to steer the optimization to simpler models.

915

## Multi-Physics Model Correction with Data-Driven Reduced Order Modelling

Stefano Riva, Carolina Introini, Antonio Cammi

Politecnico di Milano, Department of Energy, CeSNEF - Nuclear Engineering Division, Italy

stefano.riva@polimi.it

The state-of-the-art of numerical modelling for nuclear reactors is nowadays represented by the multi-physics (MP) approach. This framework enables the investigation of the inter-dependency between different physics characterising a reactor (e.g., neutronics and thermal hydraulics) for a deeper understanding of the phenomena occurring in the system. The coupling can occur in two ways: by developing interfaces between single-physics codes (e.g., Serpent for neutronics and OpenFOAM for thermal-hydraulics) or gathering every physics inside a single environment. The latter path has become state-of-the-art in the nuclear field; however, this approach loses all the previous validations of the single-physics codes; moreover, the computational resources for such complex numerical models are still demanding. Data-driven reduced Order Modelling can play a crucial role by shifting the coupling to the reduced level rather than on the full-order model, thus keeping using the already-validated and widely-used single-physics codes: the real data collected from the physical system intrinsically contain multi-physics information, and, accordingly, can be used eventually to correct the physics not considered by the model. This work applies this novel approach to a 2D coupled neutronic-thermal case study based on the PWR geometry in the Argonne National Laboratory (ANL) benchmarks; the obtained results are promising in showing the reliability and efficiency of the proposed method and paving the way for a more in-depth investigation in more complex scenarios.

916

## OpenMC Model Validation of the TRIGA Mark II Reactor

Lorenzo Loi, Carolina Introini, Antonio Cammi

Politecnico di Milano, Italy

lorenzo.loi@polimi.it

The development of open-source applications in the nuclear field has recently attracted interest from the scientific community, due to the potential mutual support useful both in the research field and in the analysis of new nuclear reactor concepts. This kind of tool allows developers to adapt the code according to their own needs, which is not generally feasible with proprietary software, designed for traditional commercial reactors and used as a black box from the user's point of view. In this framework, the OpenMC neutronic simulation tool, based on the Monte Carlo method and implemented in the accessible and easy-to-learn Python language, provides a wide range of computational capabilities, including fixed source, criticality, and burnup simulations. Verifying the reliability of open source software requires comparison processes with other already existing codes as well as benchmarks with measurement campaigns. The TRIGA Mark II research reactor is primarily used for neutron activation analysis, education, and training, making it an extensive tool for the benchmark of numerical models due to the large amount of available experimental data collected in the past years by Politecnico di Milano. In this paper, the OpenMC model validation of the current configuration of the reactor TRIGA Mark II is proposed by first comparing the control rod worth values of the three reactor control rods (REG, TRANS, SHIM); then, the experimental calibration curves of the three rods were compared with both the OpenMC and the SERPENT predictions, taking into account both the statistical error and the measurement uncertainty, aiming to demonstrate the reliability of the OpenMC code on a real reactor test-case in comparison to the widely-used SERPENT software.

917

## Feasibility study for design and utilization of a cold neutron irradiation facility at the JSI TRIGA reactor

Diego Martin Cuellar Fernandez[1], Gašper Žerovnik[1,2], Jan Malec[2]

[1] University of Ljubljana, Slovenia

[2] Institut "Jožef Stefan", Slovenia

dc5899@student.uni-lj.si

Cold neutrons range from energies of 0.1 meV to 5 meV. At these energy ranges, the cross sections for scattering and radiative capture are clearly defined outside the cross section resonance region. The need for increased accuracy in these cross section values arises from the discrepancies between different evaluated nuclear data libraries and experimental neutron cross section data.

In this regard, a design of a cold neutron source using neutrons originating from the JSI TRIGA reactor core is proposed. A parametric study of this irradiation facility is performed using computer code OpenMC for

neutron transport calculations. Neutron beam is sampled and directed through a guide tube with a cold temperature moderator at the beginning, made of graphite or H2O. At first, the simulation is done for a facility in the absence of targets, where energy spectra and cold neutron flux fractions are obtained and evaluated, using moderator thickness and temperature as degrees of freedom. Then, for selected thickness of 70 cm in graphite and 20 cm in H2O, where the neutron cold fluxes were founded to be high, the evaluation is made at different temperatures of interest and the highest cold neutron fluxes were found at around 19.6 K and 57.9 K respectively for the moderators.

With the combination of these moderator thicknesses and temperatures, different target isotopes were added and the scattering and absorption reaction rates were obtained in the simulation, leading to the calculation of their respective cross sections over the cold energy range.

---

918

## An Analytical Model of Heat Transfer in a Fuel Rod Suitable for Neutron Calculations

Muhammad Zaki Abbas Awan[1,2], Gašper Žerovnik[2], Marjan Kromar[2]

[1] Politecnico di Milano, Italy

[2] Institut "Jožef Stefan", Slovenia

muhammadzaki.awan@mail.polimi.it

In the field of nuclear engineering, understanding the response of a reactor core is crucial for ensuring safe and efficient operation. Key to this understanding are the thermohydraulic parameters of the fuel and the distribution of the neutron flux or power, as these two sets of data are intricately related. Neutron calculations, which determine the power distribution within the core, rely on temperature distributions in the fuel pellets, fuel rod jacket, and the moderator coolant. These temperature distributions, in turn, result from thermohydraulic calculations based on neutron power distributions. However, accurately determining the temperature distribution within the fuel rod presents a complex challenge due to the changing material properties of the fuel during combustion and the incomplete knowledge of all the mechanisms involved.

In this study, we propose the development of a simplified yet effective analytical model of heat transfer in a fuel rod within a nuclear reactor for neutron transport calculations. Our approach employs a one-dimensional radial geometry, where material properties remain constant along the concentric rings of the fuel rod. By assuming this simplification, we aim to achieve the necessary accuracy while minimizing computational complexity. Additionally, we will utilize an external library to incorporate dimensional and material constants essential for realistic thermohydraulic calculations, further enhancing the model's reliability.

An analytical model for accurately predicting heat transfer in a fuel rod for neutron calculations in a nuclear reactor was developed. The model’s simplicity, combined with integration of external libraries for essential constants, offers a practical and efficient solution for estimating temperature distributions within the fuel rod.

919

## Parametric Aanalysis of RBMK Fuel Depletion Calculation Using Scale Code

Sahar Shirzadi Deh Kohneh, Andrius Slavickas, Tadas Kaliatka

Lithuanian Energy Inisitute, Lithuania

sahar.shirzadi@lei.lt

Ignalina Nuclear Power Plant (INPP) is the only NPP in Lithuania with two units of RBMK-1500 shutdown in 2004 and 2009. Currently, there are about 22,000 RBMK-1500 spent fuel assemblies (SFA) at the INPP. Hence, having knowledge about the spent nuclear fuel (SNF) composition after operation in a reactor and prediction of the process in the SNF such as decay heat and activity of the radionuclides are of great importance in SNF management. SCALE code system as a comprehensive package is used for many neutronic modeling processes and SNF characterization.

In parametric analysis, two groups of parameters from T-NEWT and TRITON modules of SCALE code have been selected. Selected parameters include geometrical nodalization schemes (grids for fuel pin, fuel assembly and graphite block), parameters effecting neutron transport and depletion calculations. The objective of this study was to select a reference model based on estimated radionuclides mass sensitivities for parameter variations. The influence of the parameters is discussed for the mass of nuclides at the end of normal operation in the RBMK reactor.

The highest sensitivity to the calculation results is found for the parameter describing the number of used depletion sub-intervals in simulations (nlib) and the parameter which includes additional nuclides in the depletion scheme (addnux). The influence of other parameters on calculation results is negligible. The reference model was selected according to the results of the parametric analysis.

920

## Evaluation of the Krško NPP core using Monte Carlo approach

Dušan Čalič

Institut "Jožef Stefan", Slovenia

dusan.calic@ijs.si

The Jožef Stefan Institute (JSI) has been performing the reactor physics analysis of Krško NPP cycles for 30 years using their core simulator packages, e.g. CORD-2. This system is composed of two reactor physics codes: WIMSD-5B and GNOMER. Although WIMSD-5B is a well-known and widely used lattice code, it has limitations, as it cannot consider multiple regions in the resonance self-shielding module. To address this, in 2016, JSI explored the use of Monte Carlo transport method with the Serpent code for core design calculations to increase accuracy, despite the computational time penalty. The results were compared to the critical boron concentration and core power distribution measurements of cycle 1 of the NPP Krško, and they were satisfactory. However, the computational time was three orders of magnitude different from that of

WIMSD-5B. After some recent modifications to speed up the calculations using the Serpent and GNOMER codes the results for cycle 33 are presented. The comparison of critical boron concentrations, control rods worth, and isothermal temperature coefficient to the CORD-2 results is presented.

921

## Radiation Damage of SMR Reactor Vessel

Martin Lovecký[1], Jiří Závorka[1], Jana Jiřičková[1], Radek Škoda[1,2]

[1] University of West Bohemia, Czech Republic

[2] Czech Technical University in Prague, Czech Republic

lovecky@fel.zcu.cz

Radiation damage of reactor pressure vessel in a large reactor units is a key lifetime indicator. Number of newly proposed Small Modular Reactors (SMR) utilize a compact pressure reactor vessel that is a scaled version of larger ones. These types of SMR typically utilize nuclear fuel at a high volumetric power and along with a small reflector, resulting in fundamentally similar reactor vessel lifetime as in the large reactor units. However, there are SMR designs that operates at lower volumetric power. TEPLATOR district heating SMR is a reactor design with low pressure, low temperature and low volumetric power for district heating purposes. In this paper, radiation damage in terms of displacement per atom of TEPLATOR reactor vessel is calculated by MCNP code.

922

## An overview of research project of the nuclear power plant load follow operation

Dušan Čalič[1], Gašper Žerovnik[1], Jan Malec[1], Samo Gerkšič[1], Andrej Trkov[1], Marjan Kromar[1], Tomaž Žagar[2], Anže Mihelčič[1], Blaž Levpušček[3], Luka Snoj[1]

[1] Institut "Jožef Stefan", Slovenia

[2] Gen energija d.o.o., Slovenia

[3] University of Ljubljana, Faculty of mathematics and physics, Slovenia

dusan.calic@ijs.si

In 2023, the European Parliament and Council, on behalf of the EU members, agreed to aim for 42.5% of their energy to come from renewable sources like solar and wind by 2030. As of 2021, renewable energy accounted for 22% of the EU's total energy consumption, although this varied greatly across member states. The use of nuclear energy technology to transition towards cleaner energy systems is still unclear. Nevertheless, the latest news in April 2023 suggests that nuclear technology for nuclear-derived hydrogen is now permitted, but subject to strict industrial conditions, with further negotiations expected. Despite this uncertainty, as more renewable sources are integrated into the power grid, many countries are exploring flexible ways to operate traditional and base load energy sources such as nuclear to meet the demand for electricity and heat. This paper presents an overview of a research project funded by the Slovenian Research Agency (ARRS)

and GEN energija, which began in 2020 and focuses on load-following electricity operational mode of nuclear power plants. The paper presents the results of several work packages, including studies on calculational methodology, the simulation of flexible operation using Krško NPP, the development of a nonlinear pressurized water reactor (PWR) model, and new simultaneous independent sampling approach for sensitivity calculation, necessary to perform uncertainty propagation from nuclear data to the results of transient simulations for pressurized water reactors. The results suggest that flexible operation modes of nuclear power plants are possible, including traditional load-following mode and even some peaking capabilities with future development. Such flexible operation modes could replace the need for coal and gas-fired power plants in the future.

923

## Jožef Stefan Institute TRIGA Research Reactor Activities in the Period from September 2022 – August 2023

Anže Jazbec, Sebastjan Rupnik, Vladimir Radulović, Borut Smodiš, Luka Snoj

Institut "Jožef Stefan", Slovenia

anze.jazbec@ijs.si

The Jožef Stefan Institute (JSI) has operated a 250 kW TRIGA research reactor since 1966. Safety performance indicators (SPI) have been monitored for over ten years. Examples of the monitored parameters are; the operating time, the number of irradiated samples, doses received by operating staff, and the activity of radioactive gases released into the environment. In the paper, SPIs for the year 2022 will be presented and analysed. Such an analysis is a crucial tool to improve the future safe operation of the research reactor.

In the field of research, we continued most of the established research campaigns from the previous years. A new European project called EURO-LABS (EUROpean Laboratories for Accelerator-Based Science) was launched. It provides access to 47 research infrastructures, including the JSI TRIGA reactor.

A new experimental device is being developed. A water activation loop will be installed inside beam port no. 1 – radial piercing port. In the vicinity of the core, demineralised water will get activated. The water will be lead inside the shielded locations in the reactor hall, which can serve as a gamma radiation source of 6 MeV and 7 MeV rays. In the last year, the state of the beam port was investigated, and a detailed installation plan was developed. Most of the components are already produced and waiting to be installed. Additional concrete blocks were designed to ensure a safe working environment around the facility. The water activation device will be used as a calibration source, a practical exercise for students, and a tool to perform shielding experiments. It could also be applicable for fusion research due to high-energy gamma rays.

In the field of Education, a significant number of exercises were performed for the students from the following universities: University of Ljubljana, Uppsala University, Aix Marseille University, Politecnico di Milano and for the first time, students from King Fahd University of Petroleum & Minerals. Besides that, one EERRI and two ENEEP courses were organised where participants were young professionals in the nuclear field.

In December 2022, we organised an emergency preparedness exercise where we invited professional and volunteer firefighters from nearby stations. The scenario was an explosion during radioactive source handling inside a hot cell facility. The blast resulted in a fire and several injured workers inside the building. Including our staff, over 50 people were conducting the exercise. The paper will present the principal conclusions of the exercise.

Finally, the secondary cooling loop was replaced during winter 2022, including all pipes, valves, sensors, and heat exchangers were replaced with new items. In summer 2023, JSI plans to refurbish the purification loop for both spent fuel pools with a new pump, new pipes, and a new tank for ion exchange resins. These upgrades ensure that JSI's research reactor can continue to operate safely and efficiently for years to come.

---

924

## Computer Programs Developed at Reactor Physics Department

Bojan Žefran, Andrej Trkov, Slavko Slavič, Jan Malec, Sebastian Pleško

Institut "Jožef Stefan", Slovenia

bojan.zefran@ijs.si

At the Reactor Physics Department (F8) at the Jožef Stefan Institute (JSI) the research is focused on neutron, photon and electron transport, nuclear data evaluation and sensitivity methods. In the domain of reactor physics is directed mostly towards development of new calculational methods for research and power reactors.

One of the main activities of the department is the development of new tools - program packages that enable this reactor calculations. The Reactor Physics Department has developed several computer program packages. For example: the department's researchers have created a comprehensive package for designing a power reactor core, which has been utilized to calculate fuel cycles for the Krško Nuclear Power Plant (CORD-2). The Department has developed a new method and programs (DMR043, DMReS), known as "Rod Insertion", for measuring the value of control clusters, which significantly reduces the time required for post-refueling startup tests at nuclear power plants. The paper will briefly present the programs developed for calculations on the TRIGA reactor (TRIGLAV, The research reactor simulator (RRS), PULSTRI,...) and programs developed to perform calculations and support the operation of Krško Nuclear Power Plant (CORD-2, GNOMER, LOADF, DMRES, FAR, Shuffle,...). These programs are also commonly used for education and training. Some programs are available from Nuclear Energy Agency (NEA) Data Bank Computer Program Services and Radiation Safety Information Computational Center (RSICC), making them accessible to researchers and engineers around the world.

The paper will give an overview of the programs that have been developed at the Reactor Physics Department in the last 40 years and plans for the future.

925

## Calculation of Neutron and Electron Transport in Self-Powered Neutron Detector for PWR Fuel Assembly

Jiri Zavorka[1,2], Martin Lovecky[1,2], Radek Škoda[3]

[1] University of West Bohemia, Czech Republic

[2] ŠKODA JS a.s., Czech Republic

[3] Czech Technical University, Czech Republic

zavorka@fel.zcu.cz

The research is focused on a comprehensive study of calculating neutron and electron transport to evaluate Beta escape in a self-powered neutron detector (SPND) for a pressurized water reactor fuel assembly. The SPND is an important component used in nuclear power plants to monitor neutron flux in the core and indirect power distribution in the reactor core. Accurate neutron and electron transport calculations, considering the Beta escape phenomenon, are crucial for the proper functioning and performance assessment of the SPND.

This work presents a detailed computational model based on the Monte Carlo method to simulate the transport of neutrons, and electrons, and determination the Beta escape process within the SPND. The model incorporates the PWR fuel assembly's geometry and material properties from the perspective of neutron transport and electron transport. The primary goal of this research is to improve the fundamental knowledge in determining beta escape, i.e., the probability of an electron reaching the collector in presented model SPND. It is an essential property that specifies the sensitivity of each of the detectors of this type. Moreover, it is basically crucial for in-core monitoring.

926

## Neutronic modelling and computation of TEPLATOR core using COMSOL Multiphysics code

Dipanjan Ray, Martin Lovecký, Jiří Závorka, Radek Škoda

University of West Bohemia, Czech Republic

dipanjan@fel.zcu.cz

TEPLATOR is an innovative heavy water small modular reactor concept that has the potential to meet district and industrial heating demands by harnessing nuclear energy as a heat source. This approach offers a substantial reduction in pollution and environmental impact compared to conventional heating methods reliant on fossil fuels.

The primary objective of this article is to develop a three-dimensional, multi-group neutron diffusion model for the TEPLATOR reactor core using the COMSOL Multiphysics software package. COMSOL code employs a

finite element numerical scheme to solve the partial differential equations associated with the neutron diffusion model. Monte Carlo transport code Serpent version 2.2.1 with the latest ENDF/B-VIII.0 nuclear data library is employed to calculate the multi-group constants for different reactor regions. These multi-group constants are then incorporated in the developed COMSOL neutron diffusion model to perform calculations using adaptive mesh refinement technique to enhance the accuracy of the solution. Neutronic behaviour of the TEPLATOR reactor core is obtained by calculating criticality of the system and analysing the steady-state neutron-flux distribution profile. The calculated results subsequently compared with those generated by the Serpent, providing a basis for validation and further analysis.

927

## Development of multi-point kinetic model for TEPLATOR

Dipanjan Ray[1], Martin Lovecký[1], Jiří Závorka[1], Tomáš Peltan[1], Eva Vilímová[1], Martin Šašek[2], Radek Škoda[1]

[1] University of West Bohemia, Czech Republic

[2] ŠKODA JS a.s, Czech Republic

dipanjan@fel.zcu.cz

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 region is 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 serves as 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.

928

## Thermal Scattering Law for Zirconium Hydride

Ingrid Švajger[1], Nina C. Fleming[2], Gilles Noguere[3], Ayman I. Hawari[2], Luka Snoj[1], Andrej Trkov[1]

[1] Institut "Jožef Stefan", Slovenia

[2] North Carolina State University, USA

[3] CEA, DeS, DER, Experimental physic, Safety experiment and Instrumentation Section, France

ingrid.svajger@ijs.si

In recent years, with advances in computing power, state-of-the-art atomistic simulations based on first principles have become available for use in chemistry and physics. In these simulations, the motion of the crystal lattice or molecules is simulated using algorithms from the field of computational physics, such as density functional theory and its derivatives. The goal is to generate thermal neutron scattering cross sections and the corresponding covariance matrices. We are mainly interested in the material zirconium hydride ZrHx, which is used in the numerous research reactors like TRIGA. TRIGA reactors have a unique fuel composition: a homogeneous mixture of 20% enriched uranium and zirconium hydride (ZrH ratio close to 1.6). This is the main reason for the prompt negative temperature reactivity coefficient. Since the hydrogen in the zirconium hydride acts as a moderator, most of the moderation occurs in the fuel element itself and only a small part in the water surrounding the fuel elements. Therefore, any change in power and fuel temperature directly affects the moderator in the fuel element. In this case, the fuel and moderator directly affect the reactivity of the core. Since hydrogen bound in a crystal lattice is an essential feature of hydride fuelled reactors, accurate modelling of reactor behaviour is directly limited by the quality of thermal neutron scattering nuclear data.

ZrHx can exist in several phases with different stoichiometries, of which the δ-phase (dominant at room temperature for $1.56 < x < 1.64$) and the ε-phase (dominant at room temperature for $x > 1.74$) are the most important. Thermal neutron scattering laws for both phases are established. The system is modelled and relaxed to its ground state using the density functional theory capable computer code VASP. The positions of the atoms are then perturbed and the interatomic force constants are calculated. Once the force constants are obtained, they are transferred to the Phonopy code, which performs lattice dynamics calculations to find solutions to the dynamical matrix problem. The solutions form the dispersion relations of the system, from which the atomic vibrational density of states is computed using a geometrical sampling procedure. Once the density of states is known, the scattering law and all associated quantities can be calculated using the LEAPR module of the NJOY data processing system. The scattering law data are the basis for the analysis of thermal reactor systems, which are often modelled using Monte Carlo transport codes to obtain the desired physical reactor parameters.

The purpose of this paper is to evaluate our computational method and compare our results with the current thermal neutron scattering law data from the ENDF/B- VIII.0 library. The current ENDF/B- VIII.0 ZrHx nuclear data evaluations do not distinguish between phases. The thermal neutron scattering law data for ZrHx from the ENDF/B-VIII.0 library were generated using historical phonon spectra derived from a central force model and assume incoherent elastic scattering for both bound hydrogen and zirconium, neglecting the effects of crystal structures that are important for the scattering of zirconium bound in ZrHx. Thermal neutron scattering laws for x = 1.5 and x = 2 were established and compared with the ENDF/B- VIII.0 library.

929

## On the Dependence of Reactivity Loss With Burnup Due To Nuclear Data

Jan Malec, Andrej Trkov, Marjan Kromar

Institut "Jožef Stefan", Slovenia

andrej.trkov@ijs.si

Several new releases of evaluated nuclear data files became available recently. Published results demonstrate considerable improvement in the reactivity prediction in a broad range of criticality benchmarks. However, some concern was raised about the loss of reactivity with burnup, where the new libraries show a steeper gradient in a LWR pin model problem compared to the calculations with the ENDF/B-VII.1 or JEFF-3.1.1 libraries. The original hypothesis reported by ORNL and by UPI was that the U-238 capture data at low energies were to blame.

In order to compare the reactivity loss with burnup, a 3x3 cluster of pins of a typical PWR with 4.75 w/o enrichment operating at 1000 K was considered as an example. WIMSD5B calculations were performed with different libraries including ENDF/B-VII.0, ENDF/B-VII.1, ENDF/B-VIII.0 and JEFF-3.3. The same pin cluster was also calculated with OpenMC for the ENDF/B-VII.1 and ENDF/B-VIII.0 to verify that WIMS calculations are qualitatively correct. To draw conclusions in absolute scale, 30 cycles of the Krsko NPP were analysed with the CORD-2 system that uses WIMSD5B for pin cell calculations in a 3x3 cluster geometry. The differences in the results based on ENDF/B-VII.1 and ENDF/B-VIII.0 are compared to measured values to draw conclusions on the adequacy of new evaluated data libraries for light water reactor analysis.

930

## Evaluation of Uncertainties affecting Elastic Properties of MOX fuel

Rolando Calabrese[1], Shun Hirooka[2]

[1] Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA), Italy

[2] Japan Atomic Energy Agency (JAEA), Japan

rolando.calabrese@enea.it

Elastic properties play a relevant role in mechanical performance of nuclear fuel especially under accident conditions such as RIA and LOCA. Therefore, accurate models are mandatory for reliable fuel codes' predictions. This objective is more challenging for fast reactors where high temperatures and burn-ups lead to significant restructuring and deviations of fuel parameters from as-fabricated specifications. Results on elastic properties of MOX fuel for fast reactors published in the open literature are limited. To fill this gap measurements have been performed recently to study the effect of porosity, plutonium concentration, and deviation from stoichiometry on MOX elastic properties. Authors conducted measurements at room temperature on un-irradiated MOX fuel applying an ultrasound pulse-echo method. This paper presents a review of recent findings that aims at evaluating uncertainties affecting the elastic properties determinations published by authors. Knowledge of uncertainties allows to perform sensitivity/uncertainty analyses of codes' predictions on a sound basis.

931

## Angular Spectrum of the Diffusion Neutron Flux

Victor Kolykhanov

Odesa Polytechnic National University, Ukraine

victor.kolykhan@i.ua

The adaptation of the attenuation law of the narrow beam of particles to the attenuation of scattered diffusion flux is usually solved by using the empirical build-up factor. The main difference of the diffusion flux is the presence of a certain spectrum of angles under which the particles intersect the absorber. The process of forming a diffusion neutron flux before the intersection of the border with the absorber requires in-depth analysis.

The process of forming the diffusion neutron flux in material where going diffusion of neutrons and, in front of the boundary with the material where their absorption occurs are viewed. Was analyzed the impact of the distance from last scattering and angle of intersection the boundary.

The probability of boundary intersection the boundary was obtained and the angular spectrum of diffusion neutron flow was determined. The evolution of the angular spectrum when changing the thickness of diffusion layer and its neutronics characteristics was considered. The results obtained are the foundation for a further study of the attenuation of the diffusion neutron flow.

932

# The calorimeter design for low nuclear heating rates for the MARIA research reactor

Anna Talarowska

NCBJ, Poland

anna.talarowska@ncbj.gov.pl

The MARIA Research reactor is a neutron source for irradiation of target materials for medicine and industry: radioisotope production, Mo-99 production, neutron modification of silicon doping and minerals, Recently it regains its scientific meaning as the capabilities to design and implement irradiation rigs had been regained. In the last few years four instrumented thermostatic rigs were irradiated. Currently, there i san ongoing development of the dedicated rig for ITER diagnostic windows and discs. Crucial preparatory step is the measurement of nuclear heating in the designated irradiation channels. The problematic issue in those channels nuclear heating estimation as it has significant uncertainties. So far, the nuclear heating was estimated based on the TLD detectors irradiation and alanine detectors irradiation within one entire reactor cycle and after cool-down period the Energy deposited in the detectors was measured. Taking into account the short cycles of the MARIA Research Reactors and constantly changing core arrangement this method is far from ideal. To speed up the measurements and reduce the measurement uncertainties the calorimeter has been developed. Based on the nuclear heating measurement data the actual irradiation rigs for various samples of ITER diagnostic windows is under development. The goal of the PhD Thesis is to thermally optimize the vehicle design to meet both programatic and safety requirements of the project. The scope of the project is to gather the data which enable the informed choice of ITER diagnostic windows in the ITER facility.

# *Nuclear Fusion and Plasma Technologies*

1001

# The influence of grain size on the displacement damage creation, D retention and transport in tungsten

Sabina Markelj[1], Janez Zavašnik[1], Andreja Šestan[1], Thomas Schwarz-Selinger[2], Mitja Kelemen[1], Esther Punzon Quijorna[1], David Dellasega[3], Mateo Passoni[3]

[1] Institut "Jožef Stefan", Slovenia

[2] Max-Planck-Institut für Plasmaphysik, Germany

[3] Politecnico di Milano and Istituto per la Scienza e Tecnologia dei Plasmi, Italy

sabina.markelj@ijs.si

In a future fusion reactor neutron irradiation will create displacement damage which influences material properties, such as material strain and strength. One of the options to improve the behaviour of the material is by changing the microstructure of the material. Here we have studied how bulk microstructure, i.e. grain size in the material, influences the accumulation of radiation damage in tungsten (W). One of the hypotheses is that possibly less damage could be created due to defect annihilation at grain boundaries. To create displacement damage, we have used high energy W ions which are a good proxy for neutron irradiation, excluding transmutation, helium production and most importantly activation of the material. The amount of created defects was accessed by performing hydrogen isotope (HI) retention measurements, since lattice defects act as trapping sites for HIs with high de-trapping energy as compared to the energy of HI diffusion between solute interstitial sites. Therefore, the hydrogen isotope concentration can be treated as a measure of defect content present in the material.

The effect of microstructure on the generation of radiation damage was studied previously with W single crystal and polycrystalline W with grain sizes between 1 and 50 µm [1]. There, no significant effect was observed, with all samples showing similar D concentration in the damage zone. By using a laser-deposited, nano-crystalline W layer on a W substrate we proceeded with the study and went down with the grain size to the nanometer scale. The as-deposited layer had a grain size of few nanometers. By tempering, grain sizes of a few hundred nm up to few µm were adjusted. Samples were irradiated at 290 K by 20 MeV W-ions to create displacement damage down to 2.3 µm and with a maximum damage dose of 0.23 dpa. The concentration of defects was assessed by exposing samples to deuterium (D) atoms with an energy of 0.3 eV at 600 K and to 300 eV D ions at 450 K. In both cases D populates the created and existing defects. D retention and D depth profiles were measured by nuclear reaction analysis utilizing D(3He,p)4He nuclear reaction. In the nanograined samples D populated the damaged region more than three times faster than in samples with grain size of hundred nm and few micrometer size grains. The concentration of defects was assessed by the final D concentration in the samples. Samples with smaller grain size showed larger D concentration in the irradiated area. However, large D concentration in the non-irradiated sample showed that defect density was already high in the initial material. Samples were also analysed by transmission electron microscopy to analyse the damage distribution in the material where nanometer-size voids were observed.

[1] Pečovnik et al. J. Nucl. Mater 513, 198 (2019)

1002

# Licensing fusion facilities based on the ITER and DEMO paradigms

Andreas Ikonomopoulos[1], Mitja Uršič[2], Leon Cizelj[2], Juan Carlos de la Rosa Blul[3], Spyros Andronopoulos[1], Nicholas Terranova[4], Gian-Luigi Fiorini[5], Sigitas Rimkevičius[6], Egidijus Urbonavičius[6], Ismo Karppinen[7]

[1] National Centre for Scientific Research "Demokritos", Greece

[2] Institut "Jožef Stefan", Slovenia

[3] European Commission Joint Research Centre

[4] Italian National Agency for New Technologies, Energy and Sustainable Economic Development, Italy

[5] LISTO BVBA, Belgium

[6] Lithuanian Energy Institute, Lithuania

[7] VTT Technical Research Centre of Finland Ltd, Finland

anikon@ipta.demokritos.gr

The safety assessment for an authorized facility is performed in order to evaluate its compliance with the set of safety requirements and provide the evidence of achieving the safety objectives. It is prepared by the licensee as a structured document – safety case whose content depends on the national legislation and the nature, purpose and development phase of the facility. With regard to the DEMOnstration Power Plant (DEMO) and ITER facilities, the objective of the safety case documentation is to demonstrate the achievement of six, top-level safety objectives. [1]

To this end, three questions were posed – within the HARMONISE [2] project – while reviewing the available safety case documentation of the ITER and DEMO facilities from the point-of-view of the licensing process: What are the important outcomes that ought to be considered in the future licensing process of fusion power plants?

What improvements could be introduced in the future licensing process of fusion power plants?

What R&D tasks are required to support the future licensing process of fusion power plants?

The review performed has taken as a basis the IAEA GSR Part 4 [3] requirements with the intention to identify constituents that will contribute towards the licensing harmonization of fusion power plants in the future. As such, the facilities under consideration – an experimental and a demonstration infrastructures – represent successive implementations towards a fully operational fusion power plant. Because of their radioactive contents both ITER and DEMO are characterized as facilities in need of an authorization from a regulatory body. While the first has received it from the French nuclear safety regulator ASN (Autorité de sûreté nucléaire), the latter being in the conceptualization phase has yet to be submitted to the licensing process that will be formulated in accordance with the regulatory framework of the hosting State.

In agreement with IAEA GSR Part 4 that dictates: "A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out …" the safety cases were assessed for the two facilities at the principle level considered during designing the safety architecture of each facility with a level of detail that depends upon the information available. The different maturity levels of the two facilities provided a higher level of detail for ITER than that of DEMO.

The findings of the assessment that are based on specific designs using deuterium-tritium plasmas shaped and confined by magnetic forces will be presented and discussed outlining also the gaps that are to be filled in harmonizing the licensing process of fusion power plants in the future.

ACKNOWLEDGEMENT: This project has received funding from the European Commission – Euratom under Grant Agreement No 101061643. Views and opinions expressed are however those of the authors only and do not necessarily reflect those of the European Union or European Commission – Euratom. Neither the European Union nor the granting authority can be held responsible for them.

[1] Neill Taylor et al., Safety and environment studies for a European DEMO design concept, Fusion Engineering and Design, Volume 146, Part A, 2019, pp. 111-114
[2] https://harmonise-project.eu/, retrieved on 29 May 2023
[3] IAEA, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016)

1003

## Experimental simulation of fusion-relevant radiation environments in bulk samples

Vladimir Krsjak, Pavol Noga, Stanislav Sojak, Yamin Song, Jarmila Degmova

Slovak University of Technology, Slovak Republic

vladimir.krsjak@stuba.sk

In radiation environments of some advanced nuclear facilities such as fission, fusion, and accelerator-driven, radiation-tolerant materials may suffer from radiation embrittlement, including volumetric cavity swelling. From the material engineering point of view, it is critical to know when the microstructure of irradiated material gets to the stage where it may pose a safety hazard. Predicting the transition from safe to unsafe conditions requires understanding the early stage of the radiation ageing processes, including sinking point defects and transmutation products such as hydrogen and helium. Positron annihilation spectroscopy (PAS) is a technique known for its unique sensitivity to point defects and early-stage open-volume. However, in its most common setup, it probes a bulk of tens of micrometres. At the same time, conventional ion beam irradiation experiments used in research on nuclear materials provide displacement damage of only a few microns.

This contribution reports our recent high-energy (up to 17MeV) high-fluence (5.42 x 10^19 cm-2) helium implantation experiment, which has been used to obtain a quasi-homogenous displacement damage profile up to 70 micrometres in various samples of ferritic/martensitic steels for nuclear applications. These samples were primarily characterized by two PAS techniques: positron lifetime spectroscopy and Doppler broadening spectroscopy. Comparing the results to data obtained on similar materials irradiated in spallation neutron targets confirms the feasibility of the approach. The helium bubbles are identical to those produced in spallation target environments, typically operating at elevated irradiation temperatures and with lower He concentrations compared to the present experiment. This research provides unique engineering-relevant data for fusion and spallation communities with nearly negligible induced radioactivity of the studied samples. This approach demonstrates a convenient tool for studying radiation ageing of materials for future nuclear facilities.

Acknowledgement:
The authors would like to acknowledge support from the Slovak Research and Development Agency grants No. APVV-20-0010 and the Scientific Grant Agency of the Ministry of Education, Science, Research and Sport of the Slovak Republic and the Slovak Academy of Sciences, grant number VEGA 1/0395/20.

1004

## Simulation of fusion neutron damage in tungsten and iron using high energy protons, high energy ions, high energy neutrons and fission neutrons

Olga Ogorodnikova[1], Mitja Majerle[2], Jakub Cizek[3], Peter Hruska[3], Stanislav Simakov[4], Milan Stefanik[2], Václav Zach[2]

[1] Moscow Enginering Physics Institute, Russian Federation

[2] Nuclear Physics Institute of the CAS, Czech Republic

[3] Department of Low-temperature physics, Charles University, Czech Republic

[4] Institute for Neutron Physics and Reactor Technology, Karlsruhe Institute of Technology, Germany

olga@plasma.mephi.ru

Currently, tungsten and tungsten coatings are the reference materials of the ITER divertor and DEMO reactors and the possibility of using low-activated ferrite-martensitic, RAFM, steels not only as structural materials, but also as the material of the first wall of the fusion reactor is considered. Also, these steels, together with a new generation of RAFM steels with oxide dispersion strengthened by adding Y2O3 nanoparticles, the so-called ODS steels, are considered as promising materials for fast neutron fuel cladding. One of the key ITER and, especially, DEMO issues is radiation-induced damage caused by 14 MeV (in peak) neutron irradiation and its effect on the fuel and helium retention. As a fusion neutron source does not exist yet, to simulate fusion neutron-induced damage in materials, fission neutrons and charged particles are widely used. However, it is not always clear if the mechanisms under the ion irradiation are relevant to lower dose rate and the primary knock-on atom (PKA) spectrum under neutron irradiation. On the other hand, the fusion neutron spectrum is different from that in available fission reactors. In order to simulate the fusion experimental conditions for reliable predictions of radiation damage in fusion reactors, it is necessary to establish the adequacy of the radiation damage produced by different types of irradiation. For this reason, comparison of radiation-induced defects in metals based on W, Mo and Fe produced by high-energy self-ions, protons and neutrons with different spectrum has been performed. Radiation-induced defects have been studied by well-established method of positron-annihilation lifetime-spectroscopy (PALS), transmission electron microscopy (TEM) and nuclear reaction analysis. The study of different distributions of radiation-induced vacancies and vacancy clusters of different sizes created by different types of irradiation using PALS and TEM methods allows us an experimental validation of the value of "displacement per atom" (dpa) when comparing different types of irradiation. We found a formation of the larger size of the defects with lower density in the case of irradiation with high-energy neutrons from the p(35 MeV)-Be source compared to fission neutron- and proton- irradiations. It is shown that fission neutrons do not appear to be a good surrogate for simulating radiation damage caused by thermonuclear neutrons. Fast neutrons from p-Be source or other accelerator source can be a good surrogate to simulate radiation damage caused by fusion neutrons. Energetic protons can be a surrogate to simulate fusion neutron damage in certain materials over a certain temperature range. The new experimental data together with data available from the literature are compared with the dpa theory, including molecular dynamic simulations. Second, He/dpa ratios in different neutron facilities have been compared. We show that He/dpa ratios in the facilities with the hard energy spectra (fusion like) p(35 MeV)-Be source and DEMO are one-two orders larger than in the fission ones LVR-15, HFIR and BOR60.

Since significant amounts of helium and hydrogen will accumulate in the structural materials and in the first wall and other nodes of the reactor chamber, along with a high level of radiation damage, it is necessary to simulate the retention of helium and hydrogen in radiation-induced defects. Hydrogen embrittlement and helium swelling in a fusion reactor are important issues determining the applicability of the material, and may be the reason of shortening the lifetime of reactor components. It is shown that the hydrogen retention significantly increases in the presence of radiation damage and strongly depends on the target temperature. To predict radiation damage in DEMO, the temperature distribution along the material and the temperature gradient in the normal operation regime and during ELMs should be taken into account. Methods to obtain the best approach to modelling fusion neutron damage and to bridging the gap between theory prediction of primary defect formation and long-term damage, including gaseous and solid transmutation products, as well as thermal effects (including the temperature gradient in the normal operation regime and during ELMs) are discussed taking into account the uncertainties.

References

[1] O.V. OGORODNIKOVA, M. MAJERLE, V.V. GANN, J. ČIZEK, P. HRUSKA, S. SIMAKOV, M. ŠTEFANIK, V. ZACH, 'Verification of the theory of primary radiation damage by comparison with experimental data', J. Nucl. Mater, 525 (2019) 22-31.

[2] O.V. OGORODNIKOVA, M. MAJERLE, J. ČIZEK, S. SIMAKOV, V.V. GANN, , J. KAMENIK, J. POSPISIL, M. ŠTEFANIK, M. VINS, 'Positron annihilation spectroscopy study of radiation-induced defects in W and Fe irradiated with neutrons with different spectra', Scientific Reports volume 10, Article number: 18898 (2020)

1005

## Development of a Thermo-mechanical Model for DTT PFU

Patrik Tarfila, Oriol Costa Garrido, Boštjan Končar

Institut "Jožef Stefan", Slovenia

patrik.tarfila@gmail.com

The development of a thermo-mechanical model of the divertor plasma-facing units (PFUs) for the Divertor Tokamak Test (DTT) facility [1] will be presented. A finite element based numerical simulation of the structural model of one PFU under steady-state regime will be performed, representing an operating condition relevant to the magnetic equilibrium in which DTT will operate (single null (SN) scenario). In the simulation, the pressure load and the structural temperature field obtained in previous thermohydraulic simulations [2] will be employed as input loads. The thermo-mechanical model of the PFU will be developed assuming elastic and perfectly-plastic material properties, and a mesh sensitivity will be performed to select an accurate yet computationally efficient mesh. The expected results of the simulation represent relevant thermo-mechanical parameters to the integrity of the PFU such as displacements, stresses and deformations. Model development and numerical simulations will be performed with the ABAQUS code.

[1] G. M. Polli, “Divertor Tokamak Testing Facility (DTT): A Test Facility for the Fusion Power Plant”, (2021) Offshore Mediterranean Conference and Exhibition, 978-88946678-0-6

[2] P. Tarfila, B. Končar, O. C. Garrido, G. Dose, F. Giorgetti, and S. Roccella, “Development of a thermohydraulic model For DTT PFU,” in NENE 2022: 31st International Conference Nuclear Energy for New Europe, (2022), pp. 1012.1.-1012.8.

1006

## Optimization of Cu melt infiltration in a W skeleton for diverter material applications.

Diana Knyzhnykova, Saša Novak, Aljaž Iveković

Institut "Jožef Stefan", Slovenia

diana.knyzhnykova@ijs.si

The successful long-term operation of a fusion reactor is dependent on the durability of materials, which must be able to withstand a wide range of temperature fluctuations, neutron bombardment, magnetic forces, etc. According to the current design, the divertor, the most thermally loaded component of the fusion reactor chamber is composed of a W monoblock attached to a water cooled CuCrZr tube which acts as a heat-sink to dissipate the heat. The superior melting point, erosion resistance, and low tritium accumulation, make tungsten (W) an ideal material for plasma-facing applications. However, constant heating and cooling can cause tungsten to become thermally fatigued, which eventually leads to a loss in ductility, formation of cracks and failure of the divertor.

A composite composed of W and copper (Cu) has been proposed to solve the problem of surface cracks caused by thermal stress, by reducing the temperature gradient between the surface armour material (W) and the cooling tube (Cu/CuCrZr) . Due to the large difference in melting points of W and Cu, fabrication of such composites is challenging using conventional metallurgical processing. Typically, a combination of powder metallurgy and molten Cu infiltrationis used to form such composites. In the first step a porous W structure is created and in the second, molten Cu is infiltrated into the W framework.

A series of W samples with varying porosities were sintered for further infiltration of copper in different conditions. To achieve pore size variation of the final sintered W samples, two W powders with different particle sizes and shapes were mixed. The effect of temperature, infiltration time, atmosphere, and W pretreatment methods on the subsequent infiltration of molten copper was investigated. The resulting W-Cu composites were evaluated in terms of microstructure and residual porosity. The penetration of molten Cu in Ar-5%H2 atmosphere yielded superior results compared to pure argon. It is better to block the passage of Cu through the lateral surfaces of the sample to prevent the gas from blocking the passage of Cu through the center of the sample. The optimal conditions were used for infiltration W lattice structures fabricated by laser-based powder bed fusion.

1007

## Novel Methods of Isotopic Separation of Lithium 6

Norbert Wegrzynowski, Tomas Martin, Thomas Bligh Scott

University of Bristol, United Kingdom

n.wegrzynowski@bristol.ac.uk

Nuclear applications often require specific isotopes of an element with the right properties. An example is lithium, which has two isotopes that play a crucial role in the nuclear industry. Lithium-7 is commonly employed as a pH controller in the cooling systems of nuclear reactors, while lithium-6 is vital for tritium production. However, the natural abundance of lithium-6 is only 7.6%, which is insufficient to achieve a high breeding rate of tritium in technically feasible breeder blanket designs. Therefore, increasing the proportion of lithium-6 in any breeder blanket concept presents a significant challenge that needs to be addressed. Prior to the operation of any commercial fusion power plant, it is necessary to establish a facility capable of producing several tons of enriched lithium-6 per year. Currently, no such facility exists.

Previously, between 1950 and 1963, the COLEX process was utilized at the Oak Ridge Y-12 NSC complex. This process involved the use of toxic mercury, which had the advantage of being a non-organic working fluid, thereby avoiding the production of toxic and/or tritiated waste through decomposition. However, the use of approximately 11,000 tons of mercury, with some being released into the atmosphere, caused significant environmental damage. Consequently, this approach is no longer a viable option for the future.

This study explores the possibility of using gallium as an alternative working fluid. Gallium exhibits similar properties to mercury, with a melting point close to room temperature. The research presents a system designed to test the separation factor of a two-phase liquid-liquid chemical exchange using gallium. Previous work has demonstrated separation factors in the range of 1.03 by electrochemically introducing lithium into liquid gallium. This paper summarizes the choices involved in designing a new separation approach and compares the performance of gallium and mercury working fluids. Additionally, a novel centrifuge method is being considered and discussed in detail.

1008

## Multi-Energy Rutherford Backscattering Spectroscopy in Channeling configuration for the analysis of defects in tungsten

Esther Punzón-Quijorna[1], Miguel Luis Crespillo Almenara[2], Gastón García López[2], Xin Jin[3], Flyura Djurabekova[3], Kai Nordlund[3], Janez Zavašnik[1], Andreja Šestan[1], Thomas Schwarz-Selinger[4], Sabina Markelj[1]

[1] Institut "Jožef Stefan", Slovenia

[2] Center for Micro Analysis of Materials (CMAM), Spain

[3] University of Helsinki, Department of Physics, Finland

[4] Max-Planck-Institut für Plasmaphysik (IPP), Germany

esther.punzon-quijorna@ijs.si

In future fusion reactors one of the main material candidates for the plasma-facing components is tungsten (W), as it shows favourable properties such as high melting temperature, high thermal conductivity and low intrinsic retention of hydrogen isotopes (HI). However, in a future tokamak environment the 14 MeV neutrons from the D-T fusion reaction will create defects in the crystal lattice, altering the material properties. In order to study the created defects in the tungsten material we have used Rutherford Backscattering Spectrometry in Channeling configuration (RBS-C). Ion channeling is a well-established method for studying material properties related to its crystal structure in particular lattice disorder and defects evolution induced by ion irradiation. For quantification of the disorder, the change of the ion yield of the backscattered ions is measured along a crystallographic direction.

With the aim of studying the defects evolution by RBS-C, (111) tungsten single crystals were irradiated with 10.8 MeV tungsten ions at two different doses (0.02 and 0.2 dpa) and two different temperatures (290 K and 800 K) to create different microstructures in the material [1]. Detailed Transmission Electron Microscopy (TEM) analysis of the samples was performed showing dislocation lines and loops of different size, depending on the irradiation dose and temperature. Multi-energy RBS-C analysis along the <111> direction with four He++ beam energies of 4.5, 4.0, 3.5, and 3.0 MeV was performed. The response of the induced structural damage signal versus analysing energy gives an important information about the extension of the defects (uncorrelated or extended defects) [2,3]. For the sample with the highest damage dose the relative disorder level extracted from these measurements increases with energy. This is interpreted as extended defects such as dislocation lines which are indeed observed by TEM. In order to interpret more quantitatively the measured RBS-C spectra simulations were made by the RBSADEC code [4]. In this case realistic defects were generated from collision cascades molecular dynamics (MD) simulations. For the low damage dose sample the simulation is in good agreement with the experimental data. However, for the high damage dose it does not give the same energy dependence. The most probable reason is that the MD simulation gives only dislocation loops but no lines in contrast to the experiment. This research is performed within the EUROfusion enabling research project DeHydroC where one of the main goals consists of developing tools to

differentiate between small and large defects and to be able to study the evolution of defects during annealing or damaging by RBS-C.

[1] Hu et al. J. Nucl. Mater. 556, 153175 (2021)
[2] L.C. Feldman et al., Academic Press, San Diego, (1982), pp. 88–116
[3] Zhang, S. et al. Phys. Rev. E 94, 043319 (2016).

1009

## Divertor Armor Damage Estimation and Dynamics of Secondary Tungsten Plasma Simulation with TOKES in ITER

Leon Bogdanović[1], Sergey Pestchanyi[2], Leon Kos[1]

[1] University of Ljubljana, Faculty of Mechanical Engineering, Slovenia

[2] Karlsruhe Institute of Technology, Germany

leon.bogdanovic@lecad.fs.uni-lj.si

The TOKES code has been extensively used for several years in specific ITER studies, as well as for JET-ILW (ITER-like wall) and the EU DEMO PFC design activities. The code is used for numerical simulations of the thermonuclear deuterium-tritium (DT) plasma dynamics in a tokamak core and in the scrape-off layer (SOL), for calculations of heat flux from SOL to the tokamak walls and for heat transport inside the solid walls. It also takes into account phase transitions of the wall material – tungsten (W), simulates the dynamics of the vaporized W in the vacuum vessel, its ionization and W-D-T plasma dynamics, including photonic radiation. The original source code of TOKES is written in Pascal and compiled in Delphi, a commercial Integrated Development Environment (IDE), under Windows on single machines. In the last year, the TOKES source code has been refactored with the open source Lazarus IDE, which uses Free Pascal as an Object Pascal dialect. Versions in Lazarus for both Windows and Linux has been developed. The Linux version also runs on the HPCFS cluster, which offers superior computing capabilities compared to single machines. This paper presents TOKES simulations of the disruptions of various energy content in the core characteristic for ITER. The divertor armor damage due to melting and vaporization has been estimated and the dynamics of the secondary tungsten (W) plasma has been investigated. It has been shown that the secondary W plasma shields the divertor armor from hot DT plasma out of the core, and this shielding drastically decrease the divertor armor damage. Simulations were carried out both with the original TOKES code in Delphi and the refactored code in Lazarus. Comparison of the results is given to confirm the correct behaviour of the refactored code.

1010

# Cross section data for the Tritium-Helium-3 nuclear reaction from 0.7 to 5.1 MeV

Mitja Kelemen[1], Sabina Markelj[1], Aleksanda Cvetinović[1], Matej Lipoglavšek[1], Miha Čekada[1], Primož Pelicon[1], Mickael Payet[2], Christian Grisolia[2]

[1] Institut "Jožef Stefan", Slovenia

[2] CEA – IRFM, France

mitja.kelemen@ijs.si

Materials used in nuclear reactors (fusion or fission) are exposed to various amounts of tritium (T). At high temperatures T can permeate into the coolant and can be consequently released in to environment. In the past some methods for T detection in materials have been developed, however many of them are destructive in their nature. Ion beam methods (IBA) are often used as non-destructive methods in material science to detect trace elements in a quantitative manner. For tritium detection the nuclear reaction analysis (NRA) is the most applicable IBA technique. In the presented work we are exploring the nuclear reactions between T and 3He. 3He induces nuclear reactions with tritium, via three nuclear reaction channels [1,2,3]:

3H + 3He → 4He + 2H + 14.32 MeV (1)
→ 4He + 1H + n+ 12.09 MeV (2)
→5He + 1H + 11.36 MeV (3)
5He → 4He + n + 0.79 MeV

This reaction was investigated using a 2 MV tandem accelerator at Jožef Stefan Institute (JSI). 3He energy ranged from 0.7 MeV to 5.1 MeV and we have detected nuclear reaction products with energies between 6.5 MeV and 9.75 MeV, on tritiated samples.

The target was manufactured at JSI with triode sputtering. Silicon wafer (100) was cowered with approximately 55 nm thick layer of titanium, which was protected with 14 nm thick layer of palladium. Samples were then exposed to 1.8 bar of T2 gas at 300°C, at French Alternative Energies and Atomic Energy Commission (CEA), to saturate the titanium layer with T. The total amount of absorbed T was measured by liquid scintillation technique after outgassing the sample at 200 °C.

The IBA measurement were performed in INSIBA experimental chamber at JSI [4]. We employed small RBS detector to measure sample layer thicknesses and a large NRA detector for detection of 1H and 2H reaction products, from decay channels (1) and (3), which we were able to resolve in the spectra. The ion beam current was measured with a measuring grid positioned in front of the sample.

With such a setup, we measured a quantitative cross sections for presented reaction at scattering angles of 125°, 135° and 155°.

In general, NRA production yield is increasing with the increase of beam energy for all measured angles and given energy range. However, after the deconvolution of the NRA signal into decay channels (1) and (3) the increase of signal is only for channel (3), while the yield for channel (1) is almost constant over the whole investigated energy range. This trend is present for all the scattering angels and only the absolute values of the cross sections are different.

[1] C. Moak, Physical Review, vol. 92, p. 383, 1953.
[2] B. Kühn and B. Schlenk, Nuclear Physics, vol. 48, p. 353, 1963.
[3] U. Nocken, U. Quast, A. Richter and G. Schrieder, Nuclear Physics A, vol. 213, p. 97–106, 1973.
[4] S. Markelj et al., Nuclear Materials and Energy, vol. 28, p. 101057, 2021.

1011

## Production of high porosity lithium hydride pellets for a solid-state fusion breeder blanket concept

Alex Martin Little[1], Hugo Dominguez-Andrade[1], Norbert Wegrzynowski[1], Tom Wallace-Smith[1,2], Mahmoud Bakr Arby[1], Yannick Verbelen[1], Tomas Martin[1], Thomas Bligh Scott[1]

[1] University of Bristol, United Kingdom

[2] Astral Systems, United Kingdom

a.m.little@bristol.ac.uk

Fusion energy may be capable of supplying much of the worlds energy needs in the long term. The reaction of choice for most concepts is deuterium-tritium (D-T) fusion, as it requires the least intense conditions to achieve. Deuterium is readily extracted from water, however the acquisition of tritium is a key obstacle to overcome – it is not found to any significant degree in nature as it has a short (12.3 year) half-life. Current demand for tritium is met by CANDU fission reactors, that produce it as a by-product, but to make the kg quantities required to generate commercial scale electricity, this will not be sufficient. Therefore, most fusion concepts expect to produce tritium in-situ, by the fission of lithium with neutrons produced by the fusion reaction.

What is unclear, however, is the best way to achieve this; both liquid and solid phases of lithium are being considered, in several different compounds and chemical forms. Liquid concepts, such as molten lithium or molten lithium-lead eutectic alloy, may have advantages by doubling as a coolant. Solid concepts, such as lithium titanate or lithium oxide, could prove to be an easier alternative for tritium gas extraction but pose additional engineering challenges with circulation.

This work investigates the production of high porosity lithium hydride (LiH) pellets, as an analogue for real concepts that would use lithium deuteride (LiD). Unlike many other compounds, this removes parasitic capture: not only does the deuterium have a very small capture cross section, when it does capture, it forms tritium anyway. Additionally, released gas will likely contain approximately equal quantities of D and T, reducing or removing the need for mixing before refueling. Thanks to the high porosity and surface area, LiH/LiD provide a favorable stage for tritium gas extraction.

A range of mixtures of LiH, lithium borohydride (LiBH4), and lithium aluminium hydride (LiAlH4) have been tested for pellet manufacture. The mixtures were pressed into pellet form, then heated such that the LiBH4 and LiAlH4 decompose, liberating hydrogen gas, and leaving a mixture of LiH and B/Al/AlB2. Pellet porosity was then analysed by XRT and BET. Results of optimal mixtures for mechanical and breeding properties are discussed.

1012

## Steady-state vs. bloby plasma source: implications on divertor

Jernej Kovačič[1], Tomaž Gyergyek[2], Stefan Costea[1], Leon Kos[1]

[1] University of Ljubljana, Faculty of Mechanical Engineering, Ljubljana

[2] University of Ljubljana, Faculty of Electrical Engineering, Slovenia

jernej.kovacic @fs.uni-lj.si

In recent years we have made several studies regarding the properties of filamentary transport in a tokamak in the direction parallel to the magnetic field. The so-called blob-filaments are expected to be the dominant cross-field transport mechanism in the future tokamaks and the parallel branch of this transport mechanism is still not well understood. While experimentally we know that most of the transport is due to filaments, modelling is almost exclusively performed by steady-state source fluid codes, which then use empirically obtained correction factors to account for kinetic effects. To deepen our knowledge on this subject, we have performed fully-kinetic simulations of a tokamak magnetic flux tube using massively-parallel particle-in-cell simulation code BIT1 with two different types of source: a constant, i.e. steady-state, source of injection of particle and energy and a pulsed source. The latter has its properties set up in a way, that the particle and energy content injected into the simulation domain in one blob-filament period is on average the same as the one injected by the steady state source in the same amount of time, with exactly the same velocity distribution function.

We have then compared the effects of blob-filament temporal shape and temperature on the properties of the plasma near the bounding wall, i.e. divertor target, and on the loads of the targets. This is not only important from the direct perspective of heating due to power loads, but also from the perspective of velocity and angle dependent quantities, such as sputtering and erosion. Since we used a fully-kinetic code, we were able to develop diagnostics for angular and velocity distribution of particles absorbed by the wall and study the plasma-material interaction.

The results show that the difference between the two modelling approaches can give substantially different results on the boundary conditions and plasma-material interactions. The developed model now provides us with an improved tool for predicting the plasma-wall interactions in the fusion devices.

1013

# Radiation Tolerant Neutron Activation Detector for Compact Inertial Electrostatically Confined Fusors

Yannick Verbelen, Mahmoud Bakr Arby, Tom Scott

University of Bristol, Interface Analysis Centre, H.H. Wills Physics Laboratory, United Kingdom

yannick.verbelen@bristol.ac.uk

Ongoing research to increase the neutron flux produced by inertial electrostatically confined fusion (IECF) devices poses challenges when such systems are used to irradiate target materials, which can lead to activation rates proportional to the neutron flux. At elevated neutron flux levels, undesired activation of engineering materials, including components of the irradiation facility itself, can be expected. Whereas the majority of such activation products have short half lives that are rarely problematic, several isotopes such as Fe-55 and Co-60 occur as a result of activation of steel, and have half lives in the order of a few years which means residual activation becomes persistent on longer time scales. Monitoring the residual radiation level after shutdown of the IECF system becomes a critical component of the safety case required for operation, and it must be demonstrated that the controlled area has acceptably low residual radiation levels to protect technical operating staff from contracting radiation doses. Past accidental exposure incidents in irradiation facilities has shown that radiation detection equipment is a weak link in the radiation control chain, an observation supported by IAEA analyses of such incidents. A residual radiation detection system tailored to the specific safety requirements of an IECF system is therefore a necessity. Presented here is the design and implementation of a radiation tolerant detection system for residual beta and gamma radiation with good neutron signal rejection. The system is linked to the shield actuation system of the prototype IECF system at the University of Bristol, and experimental results with beta and gamma check sources are presented to demonstrate the effectiveness of the proposed system.

# Why APR1000?

32nd INTERNATIONAL CONFERENCE
NUCLEAR ENERGY
FOR NEW EUROPE 2023

1

### On-Time and Within Budget, Built on Experience

TEAM KOREA has the best project management capabilities in nuclear power plant construction, proven with an annual construction of one power plant since the 1980s.
We, TEAM KOREA are now constructing four APR1400 units in Korea and two APR1400 units at Barakah in the UAE, while ensuring that management know-how is maintained for future nuclear power plant projects.

2

### Excellence in Construction on Your Doorstep

Korea's enterprises proven with outstanding construction technology ensure safe and reliable nuclear power plant projects.

3

### Reliable Proven Technology, at Home and Abroad

TEAM KOREA's APR technology manifests itself with remarkable construction and operation outcomes both domestically and overseas.
By adopting state-of-the-art design features incorporating the latest technology enhancements for nuclear safety, APR technology fully meets the latest requirements of the US NRC, IAEA, WENRA, and EUR.
For over 50 years, Korea's nuclear power plants have had distinguishable achievements regarding 'zero accidents' and 'best operational performance.'

4

### Consistent Supply Chain across the Globe

TEAM KOREA cooperates with international/domestic suppliers to secure a constant supply of equipment and components, construction workforce, nuclear fuel, maintenance service, etc. This has enabled TEAM KOREA to manage nuclear power plant projects at every step with great efficiency.
TEAM KOREA highly values cooperation and inclusion of local suppliers to build a robust nuclear industry ecosystem.

Nuclear Society of Slovenia

September 2023