NENE 2016
September 5 - 8
PORTOROŽ
SLOVENIA

Book of Abstracts

25th International Conference
Nuclear Energy for New Europe
www.nss.si/nene2016

Sponsors

Golden sponsors

Silver sponsors

Bronze sponsors

Sponsors

Co-sponsor

Book of Abstracts

International Conference
Nuclear Energy for New Europe 2016

Portorož 2016

Organizers:
Nuclear Society of Slovenia
and
Jožef Stefan Institute
Reactor Physics Department

Jamova cesta 39
SI-1000 Ljubljana
Slovenia

www.nss.si/nene2016/

On the front page: The TRIGA Mark II reactor at the Jožef Stefan Institute
reached first criticality on May 31st, 1966.
Authors of the photo: Branko Čeak/Domen Pal/Jože Maček

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

Program Committee

Luka Snoj, Slovenia, Chair

Robert Bergant, Slovenia
Helmuth Böck, Austria
Oscar Cabellos, OECD/NEA
Leon Cizelj, Slovenia
Gerard Cognet, France
Nikola Čavlina, Croatia
Marko Čepin, Slovenia
Francesco D'Auria, Italy
Christophe Destouches, France
Michel Giot, Belgium
Davor Grgić, Croatia
Pavlin Petkov Groudev, Bulgaria
Tomaž Gyergyek, Slovenia
Tim Haste, France
Akos Horvath, Hungary
Tomasz Jackowski, Poland
Igor Jenčič, Slovenia
John E. Kelly, USA
Ivan Kodeli, Slovenia
Boštjan Končar, Slovenia
Janez Krajnc, Slovenia
Božidar Krajnc, Slovenia
Zdenek Križ, Czech Republic
Matjaž Leskovar, Slovenia
Borut Mavko, Slovenia
Irena Mele, IAEA
Martin Novšak, Slovenia
Primož Pelicon, Slovenia
Dubravko Pevec, Croatia
Rainer Salomaa, Finland
Igor Simonovski, EC
Lubomir Sklenka, Czech Republic
Vladimir Slugen, Slovakia
Andrej Stritar, Slovenia
Iztok Tiselj, Slovenia
Andrej Trkov, IAEA
Eugenijus Ušpuras, Lithuania
Gonzalo Jiménez Varas, Spain
Simon Walker, UK
Tomaž Žagar, Slovenia
Nadja Železnik, Slovenia

Organizing Committee

Igor Lengar, Chair

Urška Turšič
Nina Udir
Aljaž Čufar
Vid Merljak
Vladimir Radulović
Slavko Slavič
Bojan Žefran

Young Author Award Committee

Iztok Tiselj, Slovenia, Chair

Michel Giot, Belgium
Tim Haste, France
Dubravko Pevec, Croatia

Welcome

Nuclear Society of Slovenia welcomes you at the traditional meeting of professionals from nuclear research organizations, educational institutions, nuclear utilities, industrial companies and regulatory bodies. This meeting has a long tradition as it has evolved from a national conference to a regional meeting and after a decade it has gained a true international character. This year it is already the jubilee 25th conference.

Another jubilee is celebrated this year, namely 50 years of operation of the TRIGA reactor in Ljubljana, Slovenia. The reactor reached its 1st criticality in May 1966 and over the years two generations of nuclear scientists and engineers have grown working at the reactor. During this year's conference special attention will be paid also to 50 years of the Slovenian TRIGA reactor and role of research reactors to support nuclear energy. The 2016 conference's invited lectures span from problems, related to research reactors, to advanced fuel cycle strategies as well as the program for the future fusion reactor ITER. Contributed papers cover a wide range of current developments in different fields related to nuclear industry, research, education and regulation. The opportunities and challenges for nuclear power generation will be highlighted and discussed. Professionals who recognize nuclear power's importance in securing Europe's energy and environmental future are very welcomed to attend this annual conference.

Place and time of conference

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

Address:
St. Bernardin Resort
Grand Hotel Bernardin
Obala 2
6320 Portorož

From: Monday, September 5, at 16:00
To: Thursday, September 8, at 14:00

Lectures and poster sessions will be held in the Europa Convention Halls "A" and "C" on the 12th floor of Grand Hotel Bernardin.

Registration

Registration desk opening hours:

Monday, September 5:	15:00 to 19:00
Tuesday, September 6:	8:00 to 18:00
Wednesday, September 7:	8:00 to 12:30
Thursday, September 8:	8:00 to 12:00

Social Activities

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

Welcome Reception, Monday, September 5
The Welcome Reception will start at 19:30 in Grand Garden terrace at 11th floor of the Grand Hotel Bernardin with welcome drink.

Conference Trip, Wednesday, September 7
An afternoon trip to the Postojna Cave will be organized on Wednesday afternoon. Start at 14:00 in front of the hotel Bernardin.

Conference Dinner, Wednesday, September 7
It will be held at 19:30 on the 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.

Post conference activity - Technical tour to JSI TRIGA, Thursday, September 8
A technical tour to the Jožef Stefan Institute, TRIGA reactor, Ljubljana will be organized on Thursday afternoon. Start at 14:00 in front of the hotel Bernardin. Return to hotel at 20:30.

All social activities and the TRIGA technical tour are included in the registration fee.

Conference Timetable

	Monday Sept. 5	Tuesday Sept. 6	Wednesday Sept. 7	Thursday Sept. 8
8:30 - 8:50		Invited Hamid Aït Abderrahim	Invited Helmuth Boeck	Invited Simon Pinches
8:50 - 9:10				
9:10 - 9:30		Reactor physics I	Thermal hydraulics I	Fusion
9:30 - 9:50				
9:50 - 10:10				
10:10 - 10:30			Posters with coffee break	coffee break & posters
10:30 - 10:50		coffee break & posters		Radiation and environment protection
10:50 - 11:10				
11:10 - 11:30		Reactor physics II	Thermal hydraulics II	
11:30 - 11:50				Education & training pannel discussion
11:50 - 12:10				
12:10 - 12:30				
12:30 - 14:00		lunch	lunch	lunch
14:00 - 14:20		Severe accidents	Conference trip to Postojna Cave	(Technical tour to JSI TRIGA)
14:20 - 14:40				
14:40 - 15:00				
15:00 - 15:20				
15:20 - 15:40				
15:40 - 16:00		Posters with coffee break		
16:00 - 16:20	Conference opening			
16:20 - 16:40	Invited Gilles Bignan	Probabilistic safety assessment		
16:40 - 17:00				
17:00 - 17:20	coffee break			
17:20 - 17:40	Research reactors I	Reactor operation		
17:40 - 18:00				
18:00 - 18:20				
18:20 - 18:40				
19:30 --	Welcome reception		Conference dinner	

Disclaimer
The content of abstracts published in the book of abstracts is the responsibility of the authors concerned. The organizer is not responsible for published facts and technical accuracy of the presented data. The organizer would also like to apologize for any possible errors (e.g. special characters) caused by material processing. We published abstracts that have been received by August 17, 2016.

Preliminary Conference Program

Monday, Sept. 5

16:00 **Conference opening**

Invited lecture

Chairperson: Luka Snoj

No. 102 16:20

The Key role of Research Reactors in support to the development of nuclear energy: example of the JHR Project, a new Material Testing Reactor working as a European and International Users Facility in support to Research Institutes and Nucl. Industry

Gilles Bignan - France

17:20 **Research reactors**

Chairpersons: Gilles Bignan, Helmuth Böck

No. 200 17:20

Fifty years of neutron activation analysis in Slovenia

Borut Smodiš - Slovenia

No. 201 17:40

Neutron Radiography and SSNTD's at Ljubljana Triga Research Reactor: Almost 50 years of developing the methods, facilities and of research and applications

Jožef Rant - Slovenia

No. 202 18:00

Spallation Target Design for Converting the Isfahan MNSR Reactor to an Accelerator Driven System

Mohsen Kheradmand Saadi, Kimia Mokhtari - Iran

No. 203 18:20

Laboratory of fast neutron generators of the NPI

Mitja Majerle - Czech Republic

19:30 **Welcome reception**

Tuesday, Sept. 6

Invited lecture

Chairpersons: Ivan Kodeli, Dubravko Pevec

No. 103 08:30

Role of Nuclear Energy in the Future Energy Mix and Needs for R&D in Closing the Fuel Cycle
Hamid Abderrahim - Belgium

09:10 **Reactor physics I**

Chairpersons: Ivan Kodeli, Dubravko Pevec

No. 301 09:10

I2S-LWR pressure vessel fast fluence calculations
Mario Matijević, Dubravko Pevec, Krešimir Trontl - Croatia

No. 302 09:30

Variance reduction of fusion and fission neutron transport problems using the ADVANTG hybrid code
Bor Kos, Ivan Kodeli - Slovenia

No. 303 09:50

Effects of the neutronic and thermohidraulic simplifications on the neutronic power
Nicolás Olmo-Juan, Teresa María Barrachina Celda, Rafael Miró Herrero, Gumersindo Verdú - Spain

No. 304 10:10

On-the-fly towards pure Monte-Carlo transient reactor core analysis
Antonios Mylonakis, Melpomeni Varvayanni, Nicolas Catsaros - Greece

11:10 **Reactor physics II**

Chairpersons: Christophe Destouches, Marjan Kromar

No. 305 11:10

Analysis of ARC system for gas fast reactor
Filip Osuský, Lenka Dujčíková, Stefan Cerba, Gabriel Farkas, Branislav Vrban, Jakub Lüley - Slovakia

No. 306 11:30

Delayed gamma ray modeling around activated JSI TRIGA fuel elements by R2S method
Klemen Ambrožič, Luka Snoj - Slovenia

No. 307 11:50

Generation of Transport Equivalent Multi-Group Cross Sections and Diffusion Coefficients for Neutronic Analysis
Şamil Osman Gürdal, Mehmet Tombakoglu - Turkey

No. 308 12:10

Evaluation of criticality and reaction rate experimental benchmark in spherical geometry
Tanja Kaiba, Gašper Žerovnik, Luka Snoj - Slovenia

14:00 **Severe Accidents**

Chairpersons: Gerard Cognet, Ivo Kljenak

No. 801 14:00

CoreSOAR Update of the Core Degradation State-of-the-Art Report: Status September 2016

Tim Haste, Marc Barrachin, Georges Repetto, Martin Steinbrück, Paul Bottomley - France

No. 802 14:20

Nordic collaboration: Impact of Ag and NOx compounds on the transport of ruthenium in the primary circuit of NPP in a severe accident

Teemu Kärkelä, Ivan Kajan, Unto Tapper, Leena-Sisko Johansson, Melany Gouello - Finland

No. 803 14:40

Investigation of external reactor pressure vessel cooling with ATHLET-CD

Peter Pandazis, Sebastian Weber - Germany

No. 804 15:00

Comparison and analysis of corium pool behavior in lower head modeled by MAAP (EDF version) and PROCOR (CEA) codes

Sophie Bajard, Nikolai Bakouta, Benoit Habert, Romain Le Tellier, Laurent Saas - France

No. 805 15:20

Detailed Thermal-Mechanical Modelling of Cylindrical Core Support Plate during Severe Accident in PWR

Maciej Skrzypek, Eleonora Skrzypek - Poland

15:40 **Posters I**

Research reactors

No. 204 15:40

3D model of Jožef Stefan Institute TRIGA Mark II Reactor

Anže Jazbec, Luka Snoj - Slovenia

No. 205 15:40

Coolant Temperature Measurements in the core of TRIGA Research Reactor

Romain Henry, Marko Matkovič - Slovenia

No. 206 15:40

TRANSURANUS Code Performance under Fuel Melting Conditions: the HEDL P-19 Experiment

Rolando Calabrese, Paul Van Uffelen, Arndt Schubert - Italy

No. 207 15:40

Triga Reactor Simulator

Jan Malec, Dan Toškan, Luka Snoj - Slovenia

Reactor physics

No. 309 15:40

Neutron streaming analysis and shielding determination for the Krško nuclear power plant

Bor Kos, Marjan Kromar, Žiga Štancar, Peter Klenovšek, Luka Snoj - Slovenia

No. 310 15:40

SCALE 6.1.3 and Serpent 2.1.24 criticality safety analysis of a Fukushima Daiichi-like Spent Fuel Pool

Antonio Guglielmelli, Federico Rocchi, Giacomino Bandini - Italy

No. 311 15:40

Analysis of operational history of the JSI TRIGA for the purpose of benchmarking burnup calculations

Anže Pungerčič, Luka Snoj - Slovenia

No. 312 15:40

Analysis of the primary water activation in a typical PWR

Andrej Žohar, Luka Snoj - Slovenia

No. 313 15:40

Analytic function expansion nodal method (AFEN) for solving SP3 and diffusion equations in hexagonal geometry

Mohammad Hasan Jalili Bahabadi, Ali Pazirandeh - Iran

No. 314 15:40

Validation of the ADVANTG for neutron fields in three-section concrete labyrinth experimental benchmark from Cf-252 neutron source.

Domen Kotnik - Slovenia

No. 315 15:40

Current Developments of the VVER Core Analysis Code KARATE-440

György Hegyi, András Keresztúri, Csaba Maráczy, Emese Temesvári, István Panka - Hungary

No. 316 15:40

2-D reflector modelling for VENUS-2 MOX Core Benchmark

Dušan Ćalić, Andrej Trkov - Slovenia

No. 317 15:40

3D Cartesian TRIGA reactor model quality assessment by radial power distribution

Vid Merljak, Andrej Trkov - Slovenia

No. 318 15:40

Simplified In Core Fuel Management Software for Education and Training

Erhan Şenlik, Mehmet Tombakoglu - Turkey

No. 322 15:40

Determination of the Computational Bias in Criticality Safety Validation of VVER-440/V213

Branislav Vrban, Jakub Lüley, Štefan Čerba, Filip Osuský - Slovakia

No. 323 15:40

Recent development and examples of the use of the Windows interface environment XSUN-2016 for transport and sensitivity-uncertainty analysis

Ivan Kodeli, Slavko Slavič - Slovenia

No. 324 15:40

Experimental and calculated data on criticality of uranium-zirconium hydride systems with 45% enriched uranium-235

Svyatoslav Sikorin, Siarhei Mandzik, Andrei Kuzmin, Tatsiana Hryharovich - Belarus

Reactor Operation

No. 402 15:40

Developing a New Neutron and Reactivity Monitoring System for Paks NPP
Sándor Kiss, Sándor Lipcsei, Gábor Házi, Zoltán Dezső, Tamás Parkó, István Pós, Miklós Ignits, László Hományi - Hungary

No. 404 15:40

DMReS, Digital Reacivity Meter of the new Generation
Slavko Slavič, Andrej Trkov, Bojan Žefran - Slovenia

No. 405 15:40

Non-Destructive Testing of Reactor Pressure Vessel Nozzle
Petar Mateljak - Croatia

No. 406 15:40

A feasibility study on in-core fuel management via Quantum Particle Swarm optimization
Francesca Giacobbo, Gabriele Tavelli, Antonio Cammi, Marco Cauzzi - Italy

No. 407 15:40

Possibility of nuclear cogeneration development in the region of Paks
Török Szabina, Börcsök Endre, Talamon Attila - Hungary

Thermal Hydraulics

No. 508 15:40

Thermal-hydraulic Analysis Code for Plate-type Fuel Nuclear Reactors
Duvan Castellanos Gonzalez, Pedro Carajilescov, Jose Maiorino - Brazil

No. 509 15:40

The influence of imposed gas velocity profile on wave dynamics in the simulation of vertical air-water churn flow
Matej Tekavčič, Boštjan Končar, Ivo Kljenak - Slovenia

No. 510 15:40

Analysis of the MSIV Closure Transient Simulation in APROS
Tadeja Polach, Ivica Bašić, Luka Štrubelj - Slovenia

No. 511 15:40

Spectral Element Direct Numerical Simulation of Sodium Flow Over a Backward Facing Step
Jure Oder, Jernej Urankar, Iztok Tiselj - Slovenia

No. 512 15:40

Development of Turbulent Mixing Layer in Horizontal Confined Two-Component Flow
Rok Krpan, Boštjan Končar - Slovenia

No. 513 15:40

Experiments on bubbly to slug flow transition in a vertical cylindrical tube
Matic Kunšek, Daisuke Ito, Yasushi Saito - Slovenia

No. 514 15:40

Simulation of bubbly to slug flow transition in a vertical cylindrical tube with OpenFOAM computer code
Matic Kunšek, Ivo Kljenak, Leon Cizelj - Slovenia

No. 515 15:40

Evaluation of Non-condensable Gas Effect on the Operation of Emergency Core Cooling System during LBLOCA

Seunghun Yoo, Kwang-Won Seul, Young-Seok Bang - South Korea

No. 516 15:40

Transient Response of Typical VVER Steam Generator Based on RELAP and Simplified Models

Hüseyin Ayhan, Cemal Niyazi Sökmen - Turkey

No. 517 15:40

Modeling of NEK Steam Line Break analysis in computer code Apros 6

Jure Jazbinšek, Luka Štrubelj, Klemen Debelak, Ivica Bašić - Slovenia

No. 518 15:40

Assessment of Condensation Heat Transfer Models of MARS-KS and TRACE Codes Using PASCAL Test

Kyung Won Lee, Ae-Ju Cheong, Andong Shin - South Korea

No. 519 15:40

The Effect of Tube Arrangement and Turbulence Models for Steady Flow Past Tube Bundles

Ali Tiftikci, Cemil Kocar - Turkey

No. 521 15:40

Simulation of a station blackout transient using TRACE5. Application to ATLAS facility.

Maria Lorduy, Jara Turégano Lara, Sergio Gallardo, Gumersindo Verdú - Spain

No. 523 15:40

UHS Cooling Pond Evaluation using NUREG-0693 Methodology

Davor Grgić, Nikola Čavlina, Tomislav Fancev - Croatia

No. 524 15:40

Estimation of SFDS Cask Heat-up after Blockage of Ventilation Openings

Davor Grgić, Siniša Šadek, Vesna Benčik - Croatia

Materials

No. 601 15:40

Microstructural evaluation of creep behavior in hydrided E110 cladding

Hygreeva Namburi - Czech Republic

No. 602 15:40

Frequency dependencies of electrical conductivity of silicon nanoparticles exposed to neutron flux

Elchin Huseynov - Azerbaijan

No. 603 15:40

Macroscopic Validation of the Micromechanical Model for Neutron-Irradiated Stainless Steel

Samir El Shawish, Leon Cizelj, Jeremy Hure, Benoit Tanguy - Slovenia

No. 604 15:40

Drop Test Analysis of Reinforced Concrete Disposal Container

Miha Kramar, Franc Sinur, Matija Gams - Slovenia

Severe Accidents

No. 806 15:40

Influence of Melt Pouring on Stratified Steam Explosion

Vasilij Centrih, Matjaž Leskovar - Slovenia

No. 807 15:40

Simulation of natural circulation experiment in MISTRA experimental containment facility with OpenFOAM CFD code

Boštjan Zajec, Ivo Kljenak - Slovenia

No. 808 15:40

Modelling of debris bed coolability in bottom reflooding conditions with MC3D code

Janez Kokalj, Mitja Uršič, Matjaž Leskovar - Slovenia

No. 809 15:40

Comparison of CFD and LP Codes for the Simulation of Hydrogen Combustion Experiments

Tadej Holler, Ed Komen, Ivo Kljenak - Slovenia

No. 810 15:40

Improvement of the melt relocation modelling in ATHLET-CD

Liviusz Lovasz, Sebastian Weber - Germany

No. 811 15:40

Analysis of X-Ray Images in SERENA KROTOS Experiments with Premixing Simulations

Vasilij Centrih, Matjaž Leskovar - Slovenia

No. 813 15:40

Material Influence on Ex-vessel Steam Explosion

Tomaž Skobe, Matjaž Leskovar - Slovenia

No. 814 15:40

Thermal-Hydraulic Analysis of PHWR Containment using MELCOR Code in severe accident

Sungchu Song, Seon Oh Yu - South Korea

Nuclear Fusion and Plasma Technology

No. 904 15:40

CAD data storage and access in IDAM

Marijo Telenta, Leon Kos, Robert Akers - Slovenia

No. 905 15:40

Effect of multilayer insulation on thermal loading in DEMO systems

Ingrid Vavtar, Martin Draksler, Boštjan Končar - Slovenia

No. 906 15:40

Synthesis of W-based composite as a plasma facing material

Andreja Šestan, Matej Kocen, Janez Zavašnik, Saša Novak, Petra Jenuš, Miran Čeh - Slovenia

No. 907 15:40

Tunnel probe measurements in a low-temperature magnetized plasma

Jernej Kovačič, James Paul Gunn, Tomaž Gyergyek - Slovenia

No. 908 15:40

Deuterium atom loading of self-damaged tungsten at different sample temperatures
Anže Založnik, Sabina Markelj, Thomas Schwarz-Selinger, Klaus Schmid - Slovenia

No. 909 15:40

Thermal loading of divertor cassette during maintenance conditions
Luka Klobučar, Boštjan Končar - Slovenia

No. 910 15:40

The first study of deuterium retention in tungsten simultaneously damaged by high energy W ions and loaded by D
Sabina Markelj, Anže Založnik, Thomas Schwarz-Selinger, Mitja Kelemen, Primož Vavpetič, Primož Pelicon, Etienne Hodille, Christian Grisolia - Slovenia

No. 911 15:40

Micro-NRA and micro-3HIXE with 3He microbeam on samples exposed in ASDEX Upgrade and pilot-PSI machines
Mitja Kelemen - Slovenia

No. 912 15:40

Production of prompt and delayed gamma rays in fusion reactors
Dijana Makivič, Igor Lengar - Slovenia

No. 913 15:40

Deuterium Removal from Self-ion Irradiated Tungsten by Annealing in Vacuum and Isotopic Exchange
Olga Ogorodnikova, Sabina Markelj, V.V. Efimov, Yu.M. Gasparyan - Russian Federation

Radiation and Environment Protection

No. 1004 15:40

Assessment of Spent Fuel Activity in Dose Projection Software
Matic Pirc, Borut Breznik, Primož Mlakar - Slovenia

No. 1005 15:40

Radiological consequences of potential disintegration of U tailings pile at the former Žirovski Vrh uranium mine, Slovenia
Tea Bilić-Zabric - Slovenia

No. 1006 15:40

Ionization Smoke Detectors in Slovenia – Current Status and Future Challenges
Simona Sučić, Marko Kostanjevec, Tomaž Žagar - Slovenia

Education, Public Relations and Regulatory Issues

No. 1101 15:40

Public Opinion about Nuclear Energy – Year 2016 Poll
Radko Istenič, Igor Jenčič - Slovenia

No. 1102 15:40

Energy for Children
Vesna Slapar Borišek - Slovenia

No. 1103 15:40

European Decommissioning Academy (EDA) – 2nd run
Vladimír Slugeň, Martin Hornáček, Róbert Hinca, Filip Osuský - Slovakia

---

16:20 **Probabilistic Safety Assessment**

Chairpersons: Tim Haste, Matjaž Leskovar

No. 701 16:20

Shutdown Probabilistic Safety Assessment – A Case Study for the Pressurized Water Reactor
Marko Čepin, Rudolf Prosen - Slovenia

No. 702 16:40

Challenges of external hazards assessment. ASAMPSA_E project achievements.
Mirela Nitoi, Emmanuel Raimond, Yves Guigueno - Romania

No. 703 17:00

Assessments of EP&R provisions in Europe
Nadja Železnik - Slovenia

17:20 **Reactor Operation**

Chairpersons: Nikola Čavlina, Janez Krajnc

No. 319 17:20

Simulation of the Initial NPP Krško Cycles with CASL Core Simulator - VERA-CS
Andrew T. Godfrey, Fausto Franceschini, Mohamed Ouisloumen, Marjan Kromar - USA

No. 403 17:40

Systematic Approach to Training (SAT) for the design of Nuclear Power Plant (NPP) Decommissioning Training in South Korea
Jeong Keun Kwak - South Korea

Wednesday, Sept. 7

Invited lecture

Chairpersons: Iztok Tiselj, Davor Grgič

No. 104 08:30

Five Decades of TRIGA Reactors

Helmuth Böck - Austria

09:10 **Thermal Hydraulics I**

Chairpersons: Iztok Tiselj, Davor Grgič

No. 501 09:10

Investigation of Thermal Turbulent Flow Characteristics of Wire-wrapped Fuel Pin Bundle of Sodium Cooled Fast Reactor in Lattice-Boltzmann Framework

Ali Tiftikci, Cemil Kocar - Turkey

No. 502 09:30

LOCA spectrum calculations for PWR by RELAP5 and TRACE

Andrej Prošek - Slovenia

No. 503 09:50

Prediction of low-pressure subcooled boiling with advanced interfacial area source term modelling

Ronak Thakrar, Simon Walker - United Kingdom

10:10 **Posters II**

11:10 **Thermal Hydraulics II**

Chairpersons: Michel Giot, Andrej Prošek

No. 504 11:10

Crack growth assessment in pipes under turbulent fluid mixing using an improved spectral loading approach and linear elastic fracture mechanics

Oriol Costa Garrido, Samir El Shawish, Leon Cizelj - Slovenia

No. 505 11:30

Simulation of the Experiment PKL III H2.1 with the TRACE5 Code

Jara Turégano Lara, Maria Lorduy, Sergio Gallardo, Gumersindo Verdú - Spain

No. 522 11:50

Prediction of Wall Condensation in the Presence of Non-Condensable Gases through Various Thermal-Hydraulic Codes

Erol Bicer, Yeon-Joon Choo, Seong-Su Jeon, Seung-Sin Kim, Yong-Hwy Kim, Soon-Joon Hong - South Korea

No. 507 12:10

On the discontinuity of the dissipation rate associated with the temperature variance at the fluid-solid interface for cases with conjugate heat transfer

Cedric Flageul, Sofiane Benhamadouche, Eric Lamballais, Dominique Laurence, Iztok Tiselj - Slovenia

14:00 **Conference Trip**

19:30 **Conference Dinner**

Thursday, Sept. 8

Invited lecture

Chairpersons: Tomaž Gyergyek, Igor Lengar

No. 105 08:30

The ITER Integrated Modelling Programme

Simon Pinches - France

09:10 **Nuclear Fusion and Plasma Technology**

Chairpersons: Tomaž Gyergyek, Igor Lengar

No. 901 09:10

SOLPS-ITER Dashboard

Leon Kos, Ivan Lupell, Xavier Bonnin - Slovenia

No. 902 09:30

Calculations to support JET neutron yield calibration: Effects of the neutron source anisotropy

Aljaž Čufar, Paola Batistoni, Igor Lengar, Sergey Popovichev, Luka Snoj, JET Contributors - Slovenia

No. 903 09:50

Fast online MPC for ITER plasma current and shape control

Samo Gerkšič - Slovenia

10:30 **Radiation and Environmental Protection**

Chairperson: Borut Smodiš

No. 1001 10:30

MetroERM - Metrology for Radiological Early Warning Networks in Europe

Denis Glavič Cindro, Toni Petrovič, Matjaž Vencelj, Benjamin Zorko - Slovenia

No. 1002 10:50

New Ceramic Waste Forms for High Level Radioactive Wastes

Neslihan Yaniкömer, Sinan Asal, Sevilay Haciyakupoglu, Sema Erentürk - Turkey

No. 1003 11:10

Dual track approach to strategy and planning for high level waste and spent fuel deep geological disposal

Tomaž Žagar, Leon Kegel, Matej Rupret - Slovenia

11:30 **Pannel Discussion**

Challenges in Education, Training and Knowledge Management

12:30 **Conference closure**

Technical tour JSI TRIGA

Table of Contents

102

The Key role of Research Reactors in support to the development of nuclear energy: example of the JHR Project, a new Material Testing Reactor working as a European and International Users Facility in support to Research Institutes and Nucl. Industry

Gilles Bignan
CEA France, CEN Saclay ORE/SRO, France

European Material Testing Reactors (MTR) have provided an essential support for nuclear power programs over the last 50 years within the European Community. However, the large majority of these Material Test Reactors (MTRs) are more than 50 years old, leading to the increasing probability of some shutdowns for various reasons (life-limiting factors, heavy maintenance constraints, possible new regulatory requirements…). Such a situation cannot be sustained in the long term.
On the other hand, associated with hot laboratories for the post irradiation examinations, MTRs remain key structuring research facilities for the European Research Area in the field of nuclear fission energy.
MTRs address the development and the qualification of materials and fuels under irradiation with sizes and environment conditions relevant for nuclear power plants in order to optimize and demonstrate safe operations of existing power reactors as well as to support future reactor design:
- Nuclear plants will follow a long-term trend driven by the plant life extension and management, reinforcement of the safety, waste and resource management, flexibility and economic improvement.
- In parallel to extending performance and safety for existing and power plants to come, R&D programs are taking place in order to assess and develop new reactor concepts (Generation IV reactors) that meet sustainability purposes.
- In addition, for most European countries, keeping competences alive is a strategic cross-cutting issue; developing and operating a new and up-to-date research reactor appears to be an effective way to train a new generation of scientists and engineers.

This analysis was already made during the previous decade by a thematic network of EuratomFramework Program, involving experts and industry representatives, confirming the need for a new Material Testing Reactor (MTR) in Europe.This is the genesis of the Jules Horowitz Reactor (JHR), a new Material Testing Reactor (MTR) currently under construction at CEA Cadarache research center in the south of France. It will represent a major research infrastructure for scientific studies dealing with material and fuel behavior under irradiation (and is consequently identified for this purpose within various European road maps and forums; ESFRI, SNETP…).The reactor will also contribute to medical Isotope production.

Invited lectures

103

Role of Nuclear Energy in the Future Energy Mix and Needs for R&D in Closing the Fuel Cycle

Hamid Ait Abderrahim
SCK.CEN, Av. Herrmann Debrouxlaan 40, 1160 Brussels, Belgium
annita.joos@sckcen.be

Presently, the European Union produces 30% of its electricity by Gen.II and IIInuclear reactors. This leads to the production of 2500 t/y of used fuel, containing 25 t of Plutonium, and High Level Wastes (HLW) such as 3.5 t of minor actinides (MA), namely Neptunium (Np), Americium (Am) and Curium (Cm) and 3 t of long-lived fission products (LLFPs). The used fuel reprocessing followed by the geological disposal or the direct geological disposalare today the envisaged solutions depending on national fuel cycle options and waste management policies. The Partitioning and Transmutation (P&T) has been pointed out in numerous studies as the strategy that can relax constraints on the geological disposal, and reduce the monitoring period to technological and manageable time scales. Transmutation based on critical or sub-critical fast spectrum transmuters should be evaluated, in order to assess the technical and economic feasibility of this waste management option.

After nearly twenty years of basic research funded by national programmes and EURATOM framework programmes, the research community needs to reach a position of being able to quantify indicators for decision-makers, such as the proportion of waste to be channelled to this mode of management, but also issues related to safety, radiation protection, transport, secondary wastes, costs, and scheduling.

From 2005, the research community on P&T within the EU started structuring its research towards a more integrated approach. This resulted during the FP6 into two large integrated projects namely EUROPART dealing with partitioning and EUROTRANS dealing with ADS design for transmutation, development of advanced fuel for transmutation, R&D activities related to the heavy liquid metal technology, innovative structural materials and nuclear data measurement. This approach resulted in a European strategy given in introduction based on the so-called "four building blocks" at engineering level for P&T.

The MYRRHA project contributes heavily to the third building block of this European strategy and in this paper we will focus on the ADS programme in the EU through the MYRRHA project.

In this seminar we will present the EU strategy for P&T and the status of the MYRRHA project as by End-2015 concerning the technical design, the pre-licensing and the projected implementation scenario for the realization of the MYRRHA facility.

Invited lectures

104

Five Decades of TRIGA Reactors

Helmuth Böck
Vienna University of Technology, Atominstitut, Stadionallee 2, 1020 Vienna, Austria
boeck@ati.ac.at

The concept of TRIGA (Training, Research, Isotopes, General Atomics) has been developed immediately after the Geneva Conference on Peaceful Uses of Atomic Energy in 1955. The aim of the TRIGA design was a reactor that "could be given to a bunch of high school children to play with, without any fear that they would get hurt" and it should include inherent safety features. The basis of this inherent safe feature for all TRIGA type reactors is the U-Zr-H fuel with its strong negative temperature coefficient.

During the past 50 years many different types of TRIGA fuel elements with different uranium content and different enrichment have been developed but the fuel basis remained always the same.

In the late 1950ties to the end of the 1960ties TRIGA reactors were mainly commissioned in the US and Europe while later Asian countries as well as Latin America followed. Most of these reactors were used for the formation of engineers and scientist to develop a national nuclear program, at universities for academic training or at hospitals for radioisotope production. Totally 66 TRIGA reactors have been built, some were converted from MTR type fuel to TRIGA fuel. The following decades during the 1990ties and beyond are characterized by TRIGA reactors being shut down or decommissioned due to changes in the national nuclear programs, under utilization or simply lack of funds.

In addition the US fuel return program started in 1996 put pressure on many TRIGA reactors to return any HEU fuel to the US. In many countries this program initiated TRIGA shut down processes due to reasons mentioned above. Today 38 TRIGA fueled reactors remain operational.

Presently the main concern of the TRIGA community are the continuous supply of TRIGA fuel, presently suspended due to

necessary safety and security investment at the fuel factory located at Romans, France. Other concerns are costly refurbishments due to new safety and security requirements and under utilization.
After a brief history of TRIGA reactors the paper covers the present situation of the TRIGA community and gives an outlook of problems to be solved during the next decade for further successful TRIGA operation.

Invited lectures

105

The ITER Integrated Modelling Programme

Simon Pinches
ITER Organization, ITER Organization, CS 90 046, 13067 St Paul-lez-Durance Cedex, France
Simon.Pinches@iter.org

ITER is one of the most challenging and innovative scientific projects in the world today. It is a large-scale international scientific experiment involving China, the European Union, India, Japan, Korea, Russia and the United States. The principle aim is to demonstrate the viability of fusion as an energy source and to collect the data necessary for the design and subsequent operation of the first electricity-producing fusion power plant.
ITER is based on the “tokamak” concept of magnetic confinement, in which the fusion (deuterium-tritium) fuel is contained in a toroidal vessel. The ITER reactor is designed to generate 500 MW of fusion power for periods of 300 to 500 seconds with a fusion power multiplication factor, Q, of at least 10. ITER will also aim at demonstrating long fusion power production pulses, of at least 1000 seconds, with a fusion power multiplication factor of at least 5 and, ultimately, of 1 hour duration (limited only by hardware design) when full non-inductive operation is demonstrated.
A major element of the ITER Physics Research Programme is the establishment of an integrated modelling programme, including benchmarking and validation activities. Whilst this activity is co-ordinated by the ITER Organization, it is being developed using expertise and existing co-ordination structures within the ITER Members’ fusion programmes. The overall aims of this programme are to meet the initial needs of the ITER project for more accurate predictions of ITER fusion performance and for efficient control of ITER plasmas, to support the preparation for ITER operation and, in the longer term, to provide the modelling and control tools required for the ITER exploitation phase.
Support of Plasma Operations requires a set of computationally efficient, robust, physics modelling tools that are executed systematically prior to operation for pulse validation, during the pulse for plasma control and live display, and post-pulse for comprehensive reconstruction of the plasma from the full collection of diagnostic measurements. They should capture the macroscopic behaviour of the plasma with a level of fidelity that improves as ITER operation explores the new physics domain of burning plasmas. Collectively, these modelling tools comprise the Integrated Modelling Analysis Suite (IMAS).
Support of Plasma Research requires a much more extensive set of modelling tools to be employed both prior to operation and post-operation. These tools may examine microscopic behaviour, investigate more rigorous theoretical or computational behaviour, or explore new physics. They are the primary basis for model improvement and validation. They may be applied to selected pulses, segments or time slices, and may often require significant high performance computing capabilities.
IMAS, coupled with the more extensive array of physics codes in the domestic programmes, is expected to evolve toward a more self-consistent description as the ITER Research Programme progresses. One of the first applications for prototyping the IM infrastructure and developing the tools required for pulse preparation is the capability to undertake co-simulations involving the Plasma Simulator (PS) and the Plasma Control System Simulation Platform (PCS-SP).
In this presentation, an introduction to ITER and an overview of the Integrated Modelling Programme and IMAS will be presented.
The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

200

Fifty years of neutron activation analysis in Slovenia

Borut Smodiš
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
borut.smodis@ijs.si

The TRIGA Mark II reactor of Jožef Stefan Institute became critical in the year 1966. Soon after its commissioning, the installation has become to be utilized for neutron activation analysis (NAA). Early applications were dedicated to development of radiochemical procedures for determining trace elements in the environment and in human health. Particular emphasis was devoted to studying the effects of the Idrija mining and milling activities onto the environment and man. The mine and distillation plant, the second largest in the world, had been in operation since discovery of mercury in 1490. Due to its long history of discharge, the nature of the environment, the low population mobility and the reliance on local supplies of food, it represented a challenging opportunity to study the mercury transport in the environment, and its effects on biota and man.

Along with the research on mercury, the JSI scientists focused their work on radiochemical procedures for the determination of microgram and nanogram amounts of essential and toxic elements in both organic and inorganic matrices. The numerous procedures developed were largely based on solvent extraction, ion exchange and volatilization processes, and included essential and toxic elements that were difficult to be determined. Procedures comprised either isolation of a single element or simultaneous separation of a group of elements followed by their individual isolation and subsequent measurement. Radiochemical NAA (RNAA) procedures for the determination of numerous elements were developed and successfully applied in characterizing Standard Reference Materials prepared by NBS, a predecessor of the National Institute of Standards and Technology. The quality of analytical measurements developed in the laboratory resulted in long-term collaboration with eminent international organizations producing reference materials, such as USA National Bureau of Standards (NBS) – nowadays National Institute of Standards and Technology (NIST), International Atomic Energy Agency (IAEA), EU Community Bureau of Reference – nowadays Institute for Reference Materials and Measurements (IRMM) and Japanese National Institute for Environmental Studies (NIES). Many reference materials were analysed during that time either for certification purposes or simply as contribution to international high quality data.

Simultaneously, procedures for the determination of long-lived radionuclides by combination of NAA and other radiometric methods have been developed and applied.

Along with the development of nuclear and gamma spectrometric equipment in late seventies and early eighties, INAA has attracted ever more applications, gradually replacing the RNAA procedures, whenever applicable. Further decline in the application of RNAA occurred as consequence of introduction of other modern analytical techniques.

Instrumental neutron activation analysis in its relative mode had been used as soon as the laboratory received its first Ge(Li) detector in the seventies. In the eighties, the k0 –based NAA was introduced; soon after its introduction, its potential for characterizing certified standard materials was tested and confirmed. The k0 –NAA gradually replaced the relative method of INAA, eventually resulting in its accreditation as a routine analytical tool in 2009. Nowadays, the k0 –NAA is used as primary analytical tool.

In spite of ever-growing market of new and/or improved analytical techniques for elemental analysis, neutron activation analysis still plays a significant role in the preparation of reference materials due to its many favourable features.

In the presentation, the main success stories over the years are shown, the educational aspects are outlined, and the contributions towards improved quality of analytical measurements are discussed.

201

Neutron Radiography and SSNTD's at Ljubljana Triga Research Reactor: Almost 50 years of developing the methods, facilities and of research and applications

Jožef Rant
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
joze.rant@ijs.si

The idea to introduce neutron radiography (NR) at the Ljubljana TRIGA Mark II research reactor was put forward in 1968 by the present author. First NR facility (neutron collimator with beam filter, beam shutter and shielded exposure room) was constructed in the tangential beam tube and first neutron radiographs were produced in summer 1969. Later the NR facility was removed to thermal column and it was in operation until the spring 2015 when an irradiation experiment was installed in the thermal column. Already at the beginning it was a demanding challenge to develop a microneutronography (MNR) as a high resolution (~ 10 µm) neutron radiography using rather thin (few µm) Gd metal converter screens and fine grained Kodak Maximum Resolution photographic plates as a complementary method to conventional X-ray microradiography for the inspection and characterization of thin metallurgical or geological samples . Film based MNR requires high neutron exposure fluences (1010 – 1012n/cm2) and later a vertical neutron beam was constructed through the reactor water tank down to the reactor core which enabled much shorter and practical exposure times in comparison with the exposures in the tangential neutron beam. The vertical beam tube was also used for neutron induced autoradiography (NIAR or NCAR- neutron capture autoradiography) with solid state nuclear track detectors (SSNTD's) which were introduced in 1974 by the present author as a fallout of the 1973 BNES Birmingham conference and with the support of R. Barbalat and G. Farny (CEA CEN, Saclay) and of J. Barbier (Kodak Pathe'). The introduction of SSNTD's (track etch techniques) was a success and later significant research work and applications were conducted in the newly established Laboratory for SSNTD's under the leadership of R. Ilić. Early applications of MNR and NIAR/NCAR in metallurgy have been reviewed. The early development of NR and NIAR, the characterization of facilities and the use of various neutron beam tubes of te Ljubljana TRIGA have been presented elsewhere. Early applications of NR and NIAR7NCAR have been reviewed in 1981. Major refurbishment of the NR facility in the thermal column was achieved in 1995 and the photoluminescent imaging plates (IP) as a highly efficient neutron image detectors were introduced with the support of J. Stade (BAM, Berlin) in 1996. Some important applications of NR include examination of deformed irradiated TRIGA fuel elements, a study of diffusion of hydrogenous liquids into porous materials and of processes of impregnation of building materials. In the last decade applications of NR were in the field of archaeology and in the preservation of cultural heritage. Detection and maping of boron in the histological samples using NCAR with SSNTD's or IP's in developing boron neutron capture therapy for cancer was a topic of many studies. A significant achievement was the development of selective autoradiography on the basis of computerized automatic analysis and characterization of nuclear tracks by Skvarč et al. A systematic and comprehensive analysis of image quality and of image transfer response functions in radiography and autoradiography with SSNTD's was performed by Ilić and Najžer. The role of back diffused beta radiation in imaging with beta-rays emitting converter screens was evaluated by Rant et al. and the possibility to use backdiffused beta-ray radiation for simple examination of surface layers was demonstrated.

Characteristic for the past research work at Ljubljana TRIGA was, that it was not a part of the mainstream of the scientific funding, it was more the result of enthusiastic endeavour of a few collaborators and was enabled through the support within the international cooperation. The experimental possibilities and versatility of the Ljubljana TRIGA reactor was fully exploited.

202

Spallation Target Design for Converting the Isfahan MNSR Reactor to an Accelerator Driven System

Mohsen Kheradmand Saadi, Kimia Mokhtari
Department of Nuclear Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran, 1477893855, Iran
mohsen.kheradmand@gmail.com

There are many research reactors around the world that have been installed from many years ago and must be decommissioned sooner or later. However, most of the initially high enriched uranium has not been exploited yet and the reactor core has much fission products as well as actinides. The reactor conversion to an Accelerator Driven System (ADS) is one of the novel ideas for minor actinide utilization and reducing the spent fuel radio toxicity. Usually, the reactor target design is considered as a first step toward ADS design. The target performance plays an important role in ADS design and is characterized by some parameters including the spallation neutrons yield, neutron energy spectrum, deposited energy in target and the angular distribution of spallation neutrons. The main objective of this study is dedicated to spallation target design for the Miniature Neutron Source Reactor (MNSR) core conversion to an ADS one. The MNSR is a pool type research reactor, which was developed by china and installed in Isfahan nuclear technology center in 1994. After more than 20 years reactor operation, the fuel burn-up could not be compensated more by adding plate shims and the present reactor core must be removed sooner or later. The sub-criticality in MNSR reactor was attained by entirely removing the top plate shims as well as control rod and installing the spallation target in interior space of guide thimble. Different state of the art targets such as Tungsten, Lead, Bismuth, and LBE have been investigated and the target parameters were evaluated using MCNPX2.6 in both proton and neutron mode. The results showed that the neutronic performance of Tungsten is somewhat greater than its competitors. However, the Tungsten has an extraordinary thermal properties and thermal analysis of these targets is suggested for further investigations.

Research reactors

203

Laboratory of fast neutron generators of the NPI

Mitja Majerle
Nuclear Physics Institute of the CAS, Řež 130, 250 68 Řež, Czech Republic
majerle@ujf.cas.cz

The Nuclear Reactions Department of the NPI operates neutrons sources with neutron energies extending up to 35 MeV. The cyclotron U-120M provides protons in the energy range of 20-36 MeV. These are directed to a thin Li foil (quasi-monoenergetic neutrons) or to a thick Be target (continuous neutron spectrum). The available neutron fluxes are up to 10^9 n/cm^2/s for QM neutrons and 10^11 n/cm^2/s for continuous neutron spectrum.

The produced neutrons are used in a wide scale of activities connected to ADS technologies and fusion. This contribution focuses on cross-section measurement and benchmarks, the measurements of the produced neutron spectra, the development of the (n,cp) chamber and online gamma measurements.

204

3D model of Jožef Stefan Institute TRIGA Mark II Reactor

Anže Jazbec, Luka Snoj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
anze.jazbec@ijs.si

Computer assisted design (CAD) methods have already become standard in the area of engineering. However the geometrical data about the majority of the research reactor is still in the form of simple drawings and blueprints. In addition computational modelling of the reactor geometry for the purpose of neutron transport calculations or thermal hydraulics calculations still heavily relies on manual conversion of blueprints into the computer format.

Recently, activities were initiated to develop a 3D model of the completer reactor including the reactor building in a 3D CAD format. The motivation for this was the need to calculate gamma and neutron dose fields across the whole reactor hall, reactor basement and possibly inside control room.

As that attenuation of neutron and gamma fields is large, therefore standard analog Monte Carlo methods would not be very efficient. We either satisfy ourselves with large deviances or run calculation for a long time. Since none of the options is acceptable, a variance reduction technique will be used. This will be achieved by calculating weight windows with the so called CADIS method implemented in the ADVANTG package [2].

In the paper, the development of the TRIGA 3D model is described and utilisation of the model discussed.

[1] S.W. Mosher et. al., ADVANTG – An Automated Variance Reduction Parameter Generator, ORNL/TM-2013/416, Oak Ridge National Laboratory, 2013.

Research reactors

205

Coolant Temperature Measurements in the core of TRIGA Research Reactor

Romain Henry, Marko Matkovič
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
romain.henry@ijs.si

TRIGA Mark II research reactor at the "Jožef Stefan" Institute in Ljubljana is an open-pool type reactor cooled by demineralised light water. It has been used in various applications such as Neutron Activation Analysis, Neutron Radiography and Tomography and also for training personnel. Substantial amount of studies have been done with regard to the neutron physics, however, only few experiments related to reactor's thermal hydraulics have been performed so far, which makes the validation of neutron physics and reactor thermal hydraulics coupling attempt very difficult. In this light, two measurement campaigns were performed.
The first one aims to describe the natural convection process in the pool of the reactor [1].The second one focuses on axial coolant temperature profile measurement within the reactor core. Indeed, axial coolant temperature profiles along the narrow water column confined with hot fuel elements were acquired during various modes of reactor operation. For this purpose, specially tailored support structure was designed and built to accommodate 10 thermocouples in a vertical column within the core. Special attention was paid to: first, shield the temperature sensors from the hot surfaces of the fuel

elements, second, keep the sensor's tips in contact with local coolant circulation, and third, generate reduced amount of activated material.

The obtained experimental results were properly analysed and compared with CFD simulations. In fact, the only measurements of-a-kind were essential as they produced unique experimental data suitable for validation of the TRIGA's reactor core CFD model, which will further on be used for coupling attempt with the neutron physics code.

[1] Henry, Romain, Matkovič, Marko, Temperature distribution in the pool of TRIGA mark II reactor during heating transient. V: 24nd International Conference Nuclear Energy for New Europe - NENE 2015, Portorož, September 14-17.

Research reactors

206

TRANSURANUS Code Performance under Fuel Melting Conditions: the HEDL P-19 Experiment

Rolando Calabrese[1], Paul Van Uffelen[2], Arndt Schubert[2]

[1]ENEA, Via Martiri di Monte Sole 4, 40129 Bologna, Italy

[2]European Commission, Joint Research Centre, Institute for Transuranium Elements, Hermann-von-Hermoltz-Platz 1, 76344 Eggenstein-Leopolshafen, Germany

rolando.calabrese@enea.it

A reliable simulation of fuel pin behaviour is challenging due to interacting phenomena such as fission gas release, fuel/cladding swelling, thermal conductivity degradation, actinides and oxygen redistribution. Under fast reactor conditions, the description of the central void formation/closure is even more complex when power rating and geometrical conditions lead to a partial melting of fuel.

The HEDL P-19 experiment was conducted in the EBR-II reactor addressing the relationship between fuel/cladding gap width and power-to-melt. The experiment was focused on the behaviour of MOX fuel rods without preconditioning irradiation. During the experiment, power was increased up to the reactor design level (62.5 MW) which was maintained for about ten minutes. Post irradiation analyses provided information regarding the axial extension of fuel melting, central void formation, columnar grain region formation, and fuel/cladding gap width.

The experiment was conducted on sixteen encapsulated pins containing MOX fuel with an enrichment in plutonium of about 25 wt.%. The diametral fuel/cladding gap width was tailored in each pin (0.086 mm to 0.250 mm) while the cladding outer diameter was either 6.35 mm or 5.84 mm. The peak linear rating reached during the HEDL P-19 experiment was estimated to be in the interval 538 - 679 W/cm.

After a review of literature, the experiment was modelled by means of the TRANSURANUS code (2015 version) aiming to assess the performance under FBR conditions in particular considering the fuel melting correlation recommended for MOX. In addition, further information about fuel restructuring such as central void formation and plutonium redistribution models have been analysed.

207

Triga Reactor Simulator

Jan Malec, Dan Toškan, Luka Snoj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
jan.malec@student.fmf.uni-lj.si

A real time TRIGA reactor simulator was developed at the Jožef Stefan Institute. It's primary goal is to help educate students and future reactor operators, especially in developing counteries with no access to a research reactor. The behaviour of the reactor simulated resembles that of a TRIGA reactor. Its output is simulated by numerically solving point kinetics equations in the 6 group approximation and is validated by analizing the simulator's output with a digital reactivity meter[1][2]. A simple thermodynamic model has been implemented to simulate negative temperature effects on reactivity. The reactor provides multiple modes of operation. In manual mode, the operator has full control of the control rods. In Automatic mode, the control positions are automatically adjusted to maintain a desired reactor power. In square wave mode, the reactor rods are partially inserted and ejected periodically and in pulse mode, a control rod can be quickly ejected to simulate a fast transient. The reactor behaviour in the pulse mode is simulated in the Fuchs-Hans approximation. A graphical user interface enables the user to operate the reactor, visualize and analyze data such as temperature, power, reactivity, and concentration of delayed neutron precursors.

References

[1] A. Trkov, Digital Reactivity Meter DMR-043, September 13, 2004, "Jožef Stefan" Institute, Ljubljana, Slovenia
[2] LENGAR, Igor, TRKOV, Andrej, KROMAR, Marjan, SNOJ, Luka. Digital meter of reactivity for use during zero-power physics tests at the Krško NPP ISSN 1855-5748. [Tiskana izd.], feb. 2012

301

I2S-LWR pressure vessel fast fluence calculations

Mario Matijević, Dubravko Pevec, Krešimir Trontl
University of Zagreb, Faculty of Electrical Engineering and Computing, Unska 3, 10000 Zagreb, Croatia
mario.matijevic@fer.hr

The I2S-LWR concept (Integral Inherently Safe Light Water Reactor) is a high-power (1000 MWe) LWR with improved inherent (passive) safety features. This new reactor concept, led by Georgia Institute of Technology (USA), is based on the integral primary circuit configuration, a new type of the fuel/cladding system, and a novel steam generation system. This paper presents shielding studies of the I2S-LWR reactor model using SCALE6.1 code package to identify the fast neutron fluence rate distribution. The reactor pressure vessel (RPV) fast fluence calculation with uniform fission source showed that significant fast fluence of 2e19 n/cm2 was not reached, so risk from pressurized thermal shock at RPV is not impacting the reactor design for operating lifetime of 100 years. The SCALE6.1/MAVRIC shielding sequence was used to optimize neutron fluence results in the complete RPV for energies E > 0.1 MeV, E > 1.0 MeV and for the complete neutron spectra. The CADIS and FW-CADIS methodologies, based on forward-adjoint discrete ordinates (SN) solution via Denovo solver, were used to accelerate the final Monte Carlo calculation with Monaco code. Denovo utilizes Koch-Baker-Alcouffe parallel transport sweep algorithm over the XYZ meshes covering the problem domain and Krylov iteration on multigroup equations giving space-energy dependent fluxes. Such hybrid shielding methodology with mesh-based variance reduction parameters is very efficient for complex shielding problems, where particle flux is attenuated by many orders of magnitude. The same shielding methodology was used in the second part of the paper, where we calculated the radial neutron reflector heating and RPV power-level monitor's feasibility. The MAVRIC/FW-CADIS was successfully used to produce well converged neutron fluence rate (in fast and thermal region) over the reduced I2S-LWR model, extending from the core to the biological shield exterior. Obtained results were then used to optimize calculations involving inelastic neutron scattering on 28Si and 12C, since these isotopes comprise the SiC type detector which will be used for power level monitoring. Visualization of the obtained results in 3D was done using VisIt code from the Lawrence Livermore National Laboratory.

Reactor physics

302

Variance reduction of fusion and fission neutron transport problems using the ADVANTG hybrid code

Bor Kos, Ivan Aleksander Kodeli
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
bor.kos@ijs.si

Hybrid methods in an optimal way combine the best attributes of Monte Carlo (MC) and deterministic methods. Such hybrid computational radiation transport codes thusly expand the potential for solving large, complex real-world problems. The complementary use of both methods opens the way for the simulation not achievable with “analog” Monte Carlo simulations such as deep penetration or very large and complex streaming geometries.

The Automated Variance Reduction Generator (ADVANTG) code is a MC/Deterministic Hybrid transport code developed by ORNL (Oak Ridge National Laboratory), using the Denovo deterministic neutron transport code and MCNP a widely used Monte Carlo transport code. Its approach for combining deterministic and Monte Carlo transport methods is based on the Consistent Adjoint Driven Importance Sampling (CADIS) method. The fundamental concept is to generate an approximate importance function from a fast-running deterministic adjoint calculation and use the importance map to construct

variance reduction parameters, more specifically weight window parameters, which can accelerate tally convergence in the MC simulation.
ADVANTG's reliability and consistent performance has to be tested on a variety of different example problems. The use and performance of ADVANTG on three different examples encompassing a variety of neutronics applications will be presented in this paper.
Firstly the use of ADVANTG for accelerating MC simulations of deep penetration benchmark experiments such as the NESDIP3 and JANUS1 experiments will be presented. In connection with this the importance of reducing statistical uncertainty of MC simulations when validating new nuclear data will be shown.
Secondly ADVANTG coupled with MCNP will be used to determine neutron flux and neutron dose in a very large streaming geometry - the JET tokamak. More specifically the acceleration of the simulation in accordance to the NEXP benchmark experiment. The aim of this experiment is to measure the neutron streaming through ducts and the dose rates outside of the JET Torus Hall.
Lastly the use of ADVANTG for accelerating MC simulations on a newly developed detailed model of the Krško nuclear power plant will be presented. The aim of this simulations is to determine neutron dose fields in the steam generator and reactor coolant pump cubicles where "analog" simulations are difficult because of large attenuation of neutrons between the reactor core and cubicle.

Reactor physics

303

Effects of the neutronic and thermohidraulic simplifications on the neutronic power

Nicolás Olmo-Juan[1], Teresa María Barrachina Celda[2], Rafael Miró Herrero[3], Gumersindo Verdú[4]
[1]Instituto de Seguridad Industrial, Radiofísica y Medioambiental (ISIRYM), Spain
[2]Department of Chemical and Nuclear Engineering, Polytechnic University of Valencia, Camí de Vera sn, 46022 Valencia, Spain
[3]Universitat Politecnica de Catalunya, C. Jordi Girona, 31, 08034 Barcelona, Spain
[4]Universidad Politecnica de Valencia, Departamento de Ingeniería Química y Nuclear, Camino de Vera s/n, 46022 Valencia, Spain
nioljua@iqn.upv.es

In neutronic calculations, the approximation of the Boltzman neutron transport equation by diffusion equation is widely accepted. However, there are certain cases in which this approximation does not take into account the heterogeneity of the reactor core and therefore the errors in the results are not acceptable.
In such cases, other methods to solve the Boltzman neutron transport equation have to be used as for example the Simplified Spherical Harmonics, known as SPn equations.
In this paper, the analyses of the application of SP3 approximation implemented in the neutronic code PARCS in some cases, are presented. The purpose is to analyze the influence of the homogenization process of the cross sections in the results studying the influence of the Assembly Discontinuity Factors (ADF's) in the accuracy of the SP3 approximation results.
Another study carried out using PARCS in stand-alone mode is presented in this paper. When performing calculations with coupled codes, usually the thermahydraulic model of the reactor is a simplified model in which the fuel assemblies (FA) are grouped. The procedure used to group the FA affects the accuracy of the results. To analyze this influence different cases are run in PARCS stand-alone to capture the effects of the simplification of the thermalhydraulic model using different external thermalhydraulic conditions, that is, fuel temperatures and moderator densities.

304

On-the-fly towards pure Monte-Carlo transient reactor core analysis

Antonios Mylonakis[1], Melpomeni Varvayanni[2], Nicolas Catsaros[2]
[1]National Centre for Scientific Research "Demokritos", Institute of Nuclear & Radiological Sciences & Technology, Energy & Safety, Nuclear Research Reactor Laboratory, Agia Paraskevi Attikis, P.O.Box 60037, 153 10 Athens, Greece
[2]National Center for Scientific Research "DEMOKRITOS" Institute of Nuclear and Radiological Sciences and Technology, Energy and Safety Research Reactor Laboratory, PO Box 60228, 15310 Agia Paraskevi, Attiki, Greece
mylonakis@ipta.demokritos.gr

In the field of reactor physics the transient behavior of the reactor core is mainly analyzed using deterministic algorithms. However, deterministic algorithms make use of various approximations mainly in geometric and energetic domain which may induce inaccuracy. On the other hand Monte-Carlo analysis, which generally does not require significant approximations, is currently very extensively used in static problems but not in transient analysis. Since nowadays the available computational resources are continuously increasing, the potential use of the Monte-Carlo methodology in the field of reactor transient analysis seems quite attractive. This work performs an investigation of this possibility by developing a Monte-Carlo transient solver on the open-source Monte-Carlo static code OpenMC. The obtained results are encouraging giving motivation for further investigation and development.

Reactor physics

305

Analysis of ARC system for gas fast reactor

Filip Osuský[1], Lenka Dujčíková[1], Stefan Cerba[1], Gabriel Farkas[2], Branislav Vrban[1], Jakub Lüley[1]
[1]Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Institute of Nuclear and Physical Engineering, Ilkovičova 3, 812 19 Bratislava 1, Slovakia
[2]Slovak University of Technology Faculty of Electrical Engineering and Information Technology Department of Nuclear Physics and Technology, Ilkovičova 1, 812 19 Bratislava, Slovakia
filip.osusky@stuba.sk

The paper is focused on application of assembly reactivity control (ARC) system within gas fast reactor (GFR). The ARC system provides negative reactivity feedback without damaging the neutron economy. Liquid/liquid system is used and the idea is that the separate liquid pushes 6Li in to the core region after temperature increase. Potassium is current best choice for the expansion liquid with low solubility with lithium, large thermal expansion coefficient, low neutron absorption cross-section, low corrosion with the cladding materials and is chemically stable under irradiation. The main idea is replacement of one or more fuel pins by ARC injection rods with minimal change to fuel assembly. Liquid reservoir is located in the upper part of fuel assembly with neutronically transparent liquid. The ARC injection rod consists of two concentric tubes where the inner tube is filled with potassium and outer tube with argon. The lower reservoir contains dual-layer of liquids with floating 6Li on potassium. The absorber in the form of 6Li is pushed in to the outer tube with the temperature increase by thermal expansion of potassium. Different speed of control system actuation can be achieved by changing of diameter for inner and outer tube. Second recriticality of fast reactor core is discussed based on the steady state neutronics calculations. It is assumed that the molten core is relocated within fixed core boundaries and new core compaction is responsible for second recriticality of the nuclear system. The purpose of the ARC system is to mitigate such event and to overcome the issue of too positive coolant temperature feedback and too large positive coolant void worth. The analysis provides

reactivity worth of system with different number and type of ARC rods within the fuel assembly by SCALE code. The investigated cases are during normal operation and during voiding of coolant.

Reactor physics

306

Delayed gamma ray modeling around activated JSI TRIGA fuel elements by R2S method

Klemen Ambrožič, Luka Snoj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
klemen.ambrozic@ijs.si

The Jožef Stefan Institute (JSI) TRIGA reactor is a 250 kW, pool type reactor with fuel elements arranged in an annular configuration, which is equipped with numerous irradiation facilities with well characterized neutron fields (Snoj, Žerovnik, & Trkov, 2012) and has become a reference center for neutron detector testing for Atlas experiment, CERN (Cindro, Kramberger, & Mandić, 2005).
Prompt gamma ray production can already be calculated using existing Monte Carlo particle transport codes such as MCNP (Goorley, 2012) and nuclear data libraries such as ENDF/B-VII.1 (Chadwick, 2011). However delayed gamma generation and isotopic changes are commonly not yet supported. To this end, Rigorous two-step (R2S) method codes have been developed and incorporated with different degrees of accuracy (Batistoni, Angelone, Petrizzi, & Pillon, 2002). In this article, an in-house developed R2S method code is described, and results of its application for calculation of delayed gamma-ray flux and dose from activated nuclear fuel, as well as modifications to the isotopic concentrations and their contributions to gamma dose are presented.
Activated nuclear fuel is the dominant source of delayed gamma rays in the reactor, and is commonly used as a source of gamma rays for gamma sample irradiation.
An R2S method couples Monte Carlo particle transport codes with neutron activation and transmutation codes, superimposing a 3D mesh over Monte Carlo model geometry, where neutron spectrum and total flux are calculated in each voxel of the mesh. Delayed gamma ray spectra and intensities are then calculated in all voxels using neutron activation code, and input into the Monte Carlo particle transport model as gamma sources respectively.

References:

Batistoni, P., Angelone, M., Petrizzi, P., & Pillon, M. (2002). Benchmark Experiment for the Validation of Shut Down Activation and Dose Rate in a Fusion Device. Journal of Nuclear Science and Technology, 39, 974-977. doi:10.1080/00223131.2002.10875263
Chadwick, H. O. (2011, December). ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data. Nuclear Data Sheets, 112(12), 2887-2996.
Cindro, V., Kramberger, G., & Mandić, I. (2005). Irradiation studies for the ATLAS inner detector opto-electronic readout system for SLHC. 11th Workshop on Electronics for LHC and Future Experiments (pp. 311-315). Heidelberg, Germany: CERN.
Goorley, J. T. (2012, December). Initial MCNP6 Release Overview. Nuclear Technology(180), 298-315.
Snoj, L., Žerovnik, G., & Trkov, A. (2012, March). Computational analysis of irradiation facilities at the JSI TRIGA reactor. Applied Radiation and Isotopes, 70(3), 483-488.

307

Generation of Transport Equivalent Multi-Group Cross Sections and Diffusion Coefficients for Neutronic Analysis

Şamil Osman Gürdal, Mehmet Tombakoglu
Hacettepe University, Nuclear Engineering Department, 06800 Beytepe, Ankara, Turkey
mtombak@hacettepe.edu.tr

In this study, generation of transport equivalent assembly averaged macroscopic cross section set using Monte Carlo technique is discussed for graphite and light water moderated reactors. One of the contributions of this study is demonstration of cell averaging technique to find an expression for direction dependent diffusion coefficient using the simulation results of MCNP5 code with analytical results obtained by using diffusion theory. It should be noted that, reaction rates and flux shapes obtained by using diffusion theory becomes equivalent to transport theory results for two group transport equivalent cross section set and diffusion parameters. The results are also compared with the lattice cell code results for benchmark problems defined in literature.

Reactor physics

308

Evaluation of criticality and reaction rate experimental benchmark in spherical geometry

Tanja Kaiba, Gašper Žerovnik, Luka Snoj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
tanja.kaiba@ijs.si

Critical and reactor physics experiments involving aqueous uranyl fluoride (UO2F2) solutions were performed at the Oak Ridge National Laboratory (ORNL) between 1958 and 1960. In order to determine under which conditions the aqueous solutions of intermediate enriched uranium (37 wt% 235U) can be made critical and to determine basic physical parameters. Second experimental part was performed to evaluate reaction rate distribution inside the sphere. Radial fission rate profile was measured using two fission chambers, one was used as a static counter inside the sphere, while other was positioned inside guide tube and was moving vertically through the sphere. Radial profile was measured relative to the static counter and relative to the center of the sphere. The computational model of the experiments was made in the Monte Carlo neutron transport code MCNP based on experimental reports and logbooks. The model and the MCNP code were than used to perform the evaluation of experimental uncertainties of the measured quantities, i.e. k-eff and fission rate profile, according to the methodology proposed by ICSBEP (International Criticality Safety Benchmark Evaluation Project). Different contributions to the overall uncertainty were studied, such as: uncertainties in solution volume, enrichment, uranium concentration, solution impurities, departure from sphericity, surrounding structure etc. the most important for the criticality evaluation being uncertainty in enrichment. It has been found that the experimental uncertainty is low enough to consider evaluations to be published in the ICSBEP and IRPhE (International Reactor Physics Benchmark Experiment Evaluation Project). The criticality benchmark has already been published in the ICSBEP handbook under the identifier IEU-SOL-THERM-005, while the evaluation of reaction rate measurements is still in progress.

309

Neutron streaming analysis and shielding determination for the Krško nuclear power plant

Bor Kos[1], Marjan Kromar[1], Žiga Štancar[1], Peter Klenovšek[2], Luka Snoj[1]
[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]Nuklearna elektrarna Krško, Vrbina 12, 8270 Krško, Slovenia
bor.kos@ijs.si

At the Nuclear Power Plant Krško it was decided to evaluate possibilities of minimizing neutron streaming radiation dose rates from the reactor core to reasonably low levels to reduce staff radiation exposure entering and working in the reactor building and also to reduce long term effects on radiation-sensitive equipment exposed to neutron radiation. This action is particularly important for locations where significant neutron streaming is present. The purpose of the shield determination is to reduce the neutron dose for personnel in SG (steam generator) and RCP (reactor coolant pump) cubicles during reactor operations low as reasonable achievable.
A detailed geometrical model of the Krško nuclear power plant for Monte Carlo neutron transport calculation was made based on CAD (computer assisted design) models, blueprints, technical drawings and other available data. Monte Carlo simulations to determine the absolute and relative dose fields are performed with the general purposes Monte Carlo neutron transport code MCNP. Monte Carlo calculations are coupled with deterministic neutron transport codes to determine optimal variance reduction parameters, such as cell importances.
In the paper basic (conceptual) design of shields placed at locations around RCS (reactor coolant system) hot and cold leg piping entering SG and RCP cubicles to reduce neutrons streaming through the RCS loop pipe penetrations is presented. In order to determine optimal position and dimensions of the shield, a thorough parametric analysis is performed including assessing the importance of neutron streaming through the reflective insulation of the RCS piping. A conceptual and physics analysis of shielding is made in order to enhance understanding in the neutron transport from the reactor core to the cubicle and to determine the neutron paths and most important components from neutronic point of view.

Reactor physics

310

SCALE 6.1.3 and Serpent 2.1.24 criticality safety analysis of a Fukushima Daiichi-like Spent Fuel Pool

Antonio Guglielmelli[1], Federico Rocchi[2], Giacomino Bandini[2]
[1]Italian National Agency for New Technology, Energy and Substainable Economic Development, Via Martiri di Monte Sole, 4 - Bologna, 40129, Italy
[2]ENEA, Via Martiri di Monte Sole 4, 40129 Bologna, Italy
antonio.guglielmelli@gmail.com

The Fukushima Daiichi nuclear power plant accident has highlighted the risk of criticality safety problems in the spent fuel pools (SFPs) used to store fresh and/or burnt fuel assemblies of a nuclear reactor under specific conditions. In a SFP the fuel arrangement and the geometrical configuration must be designed to keep such a system with a given subcriticality margin to ensure criticality safety under both operational and credible accidental conditions. In the framework of the Nugenia+ AIR-SFP Project, and with the aim to verify the existence of the proper critical safety margin under accidental conditions in a SFP similar to that of the Fukushima Daiichi Unit 4, a series of calculations have been executed with the

Montecarlo code KENO VI of SCALE 6.1.3 package by means of both continuous-energy and multigroup cross sections based on the ENDF/B-VII.0 library. The criticality simulations have been performed both on a single unit-cell of a rack and on a whole 3x10 SFP rack. The fuel assembly considered was a fresh 9x9-9 BWR FA equipped with 12 Gd-doped pins whose material and geometrical description have been taken from the specifications of the OECD/NEA Burn-up Credit Criticality Benchmark Phase IIIC. Preliminarily, the criticality of the initial safe state (isothermal @ 25 °C , water density 1 g/cm3) and that of accidental conditions with uniform properties (isothermal @ 100 °C, water density between 1 and 0.1 g/cm3); have been evaluated. Subsequently - employing the results of some RELAP5 thermo-hydraulic calculations – the criticality safety margin for more realistic accidental conditions (non-isothermal fuel, water density distribution) has been estimated. The thermo-hydraulic simulations have been achieved assuming a loss of coolant accident (LOCA) with a partial fuel uncover, a “rod bundle” correlation, and a fuel decay heat of 9.0 kW corresponding to a decay of about 15 days after shutdown and at burn-up of 12 GWd/MTU. The effect of a burn-up of 12 GWd/MTU and of the corresponding depletion of gadolinium on the criticality margin has been taken into account a-posteriori using a lumped, pre-estimated reactivity increase. Finally, it has also been realized a sensitivity analysis to estimate the effect on criticality of different rack unit interspacing.

Reactor physics

311

Analysis of operational history of the JSI TRIGA for the purpose of benchmarking burnup calculations

Anže Pungerčič, Luka Snoj

Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia

anze.pungercic@student.fmf.uni-lj.si

The TRIGA reactor at the JSI started operating on 31st May 1966. Since then 320 different fuel elements were used, arranged into 218 reactor cores. Our goal is to simulate 50 years of operation by using deterministic (TRIGLAvW) and stochastic (SERPENT) neutron transport and burnup codes and validate the calculations by experimental and operational data. In order to perform this a complete reactor operational history had to be analysed and put into a computer readable format. First important quantity, directly connected with burnup through energy released from fission of a nucleus (e.g. Uranium-235), is the thermal energy generated in the reactor. This data is stored in reactor operation logbooks, which are written by hand, therefore digitalization is required. The logbooks also contain information regarding the fuel shuffling between different reactor cores and excess reactivity, which is measured every Monday since the beginning. Fuel element burnup could be determined with well-known methods: reactor calculations, gamma ray spectrometry of irradiated fuel measurement of the elements relative reactivity worth. Last two methods cannot be performed for majority of the fuel elements as more than 200 elements were shipped back to USA in 1999.

The main purpose of this paper is to describe the analysis of all information required for burnup simulations and show the quantity of reactor cores used in calculations with Monte Carlo. The fuel element burnup accumulated during 1966-2016 will be calculated with two codes; the TRIGLAV fuel management two-dimensional multigroup diffusion code and the Monte Carlo neutron transport code SERPENT. Burnup modules in both codes will be compared against each other and calculated excess reactivities will be compared against the measurements.

312

Analysis of the primary water activation in a typical PWR

Andrej Žohar, Luka Snoj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
andrej.zohar@student.fmf.uni-lj.si

In pressurised water reactors the cooling water in primary loop is radioactive due to activation of the water itself, activation of corrosion products, migration of fission and activation products through the cladding. We are going to analyse activation and activity of water in the model of a typical two loop PWR nuclear power plant.
We will focus mostly on the most important nuclides, i.e. activation of different oxygen nuclides, especially O-16, due to high natural abundance (99.76%) and high energy gamma radiation of activated product N-16 (~ 6 and 7 MeV). Other important oxygen nuclides are O-17 and O-18, due to high energy neutrons from N-17 decay (~ 1 MeV) and high energy gamma from O-19 decay (~ 0.2 and 1.4 MeV).
The neutron spectra and reaction rate calculations is performed using the Monte Carlo neutron transport code MCNP. In parallel water activation is calculated by using activation codes FISPACT and ACAB. The calculations will be performed with various up to date nuclear data libraries, such as ENDF/B-VII.0, TENDL-2015, JEFF-3.2 and EAF-2010. It can be observed that some nuclear data libraries lack the cross section energy dependence for activation of O-18. Among the previously mentioned nuclear data libraries, library ENDF/B-VII.0 lacks this cross section energy dependence. In addition, some cross section energy dependences from different libraries differ significantly between each other, especially for the activation of O-18.
The neutron spectrum calculations are performed for hot zero power conditions at various positions inside the reactor pressure vessel.
Calculations of the time dependence of activity for each presented activated nuclide will also be carried out. The time dependence of activity is going to include behaviour at the start of the reactor, behaviour at changes of power of the reactor and behaviour after shutdown of reactor. The results will then serve to describe a gamma ray source in subsequent Monte Carlo photon transport calculations to evaluate dose fields around the primary loop.

Reactor physics

313

Analytic function expansion nodal method (AFEN) for solving SP3 and diffusion equations in hexagonal geometry

Mohammad Hasan Jalili Bahabadi[1], Ali Pazirandeh[2]
[1]Department of Nuclear Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran, 1477893855, Iran
[2]Islamic Azad University, Science and Research Campus Islamic Azad University, Hesarak St, Ashrafi Isfahani Boulevard, Tehran 1454696111, Iran
jalili.mohammadhasan@gmail.com

In this paper, two nuclear codes named MGHANSP3 and HexDANM are introduced. In MGHANSP3 and HexDANM codes, the AFEN method was utilized to solve SP3 and diffusion equations in hexagonal geometry respectively. This method represents a multidimensional intra nodal flux distribution in terms of analytic basis functions at any points in the node. The surface averaged partial currents in half of the surfaces of the hexagon adopted at the nodal boundary coupling conditions. Finally, the IAEA benchmark problem was used to comparison of SP3 and diffusion theory. The numerical results show that the MGHANSP3 code more accurate and also slower than HexDANM.

Reactor physics

314

Validation of the ADVANTG for neutron fields in three-section concrete labyrinth experimental benchmark from Cf-252 neutron source.

Domen Kotnik
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
domen.kotnik@student.fmf.uni-lj.si

ADVANTG, Automated Variance Reduction Generator [1], is a code developed by the Oak Ridge National Laboratory that aims to automate the process of generating variance reduction parameters for fixed source MCNP calculations. As it was released recently it has not been tested on majority of available experimental benchmarks.

The purpose of this paper is to validate the use of ADVANTG on the ICSBEP (International Criticality Safety Benchmark Evaluation Project) shielding benchmark ALARM-CF-AIR-LAB-001, i.e. neutron fields in three-section concrete labyrinth from Cf-252 neutron source [2].

Experimental investigation of the neutron flux in a large three-section concrete labyrinth was done in the summer of 1982 in an open area at the Institute of High Energy Physics at Protvino, near Serpukhov (Moscow Region), Russia. The source of neutrons for these experiments was spontaneous decay of 252Cf. The source was installed at the center of the doorway aperture of the labyrinth. The experiments were performed with “unfiltered” radiation from the bare source and with “filtered” radiation from the source surrounded by a 30.5-cm-diameter polyethylene sphere with a 4-cm-diameter spherical central cavity. Neutron flux was measured by the Bonner sphere method at different points inside each section of the labyrinth. The influence of different coverings of the labyrinth wall on the neutron flux in remote sections of the labyrinth was investigated. The aim of the experiments was to obtain benchmark data for validation of the computer codes used for estimation of doses from the neutrons that penetrate the shielding, via numerous leaks, of the acceleration-storage ring of the Large Serpukhov Proton Accelerator.

The calculations are divided in 2-steps process because of the geometry of the labyrinth and consequently extremely low number of 6Li (n, α) reactions in the detectors crystal. First step is calculation of the response function (sensitivity) of the detectors (Bonner spheres) and then calculation of the neutron flux at points of the measurement positions.

Results from a MCNP calculations fits very well with the experiment. In order to get statistical reliable results MCNP need long computational time because of the extremely low number of neutrons which toward to detectors reactions. Hence ADVANTG is used to speed up calculation.

References:

[1] S.W. Mosher, A.M. Bevill, S.R. Johnson, A.M. Ibrahim, C.R. Daily, T.M. Evans, J.C. Wagner, J.O. Johnson, R.E. Grove. 2013. ADVANTG-An Automated Variance Reduction Parameter Generator. Oak Ridge National Laboratory, Oak Ridge.

[2] E.A. Belogorlov et al. 1984. Neutron fields investigation in three sectional concrete labyrinth from Cf-252 source, ALARM-CF-AIR-LAB-001. Serpukhov, IHEP.

315

Current Developments of the VVER Core Analysis Code KARATE-440

György Hegyi[1], András Keresztúri[2], Csaba Maráczy[2], Emese Temesvári[2], István Panka[3]
[1]Retired, Hungary
[2]Centre for Energy Research Hungarian Academy of Sciences, P.O.Box 49, H-1525 Budapest, Hungary
[3]Hungarian Academy of Sciences Centre for Energy Research, Budapest 114, P.O. Box 49, Hungary, H-1525, Hungary
gyorgy.hegyi@energia.mta.hu

Due to the new challenges (more heterogeneous and higher enriched fuel assembly, the safety requirements) the updating of nodal codes for steady state and transient core analysis is carried on continuously to enhance the accuracy and robustness. The fuel modifications and the upgraded regimes requiring more accurate calculations have necessitated the further development and validation of the KARATE code system.
On the other hand even the calculations have reached a high quality level; it is very important to take into account the uncertainties of the calculations, especially the uncertainties of the input parameters related to the applied models which cannot be eliminated. A realistic estimation of these uncertainties is necessary for judging the reliability of the simulation results. Recently there is a tendency to use best estimate plus uncertainty methods in the field of nuclear energy. This implies the application of best estimate code systems and the determination of the appropriate uncertainties.
Taking account of these goals, the following improvements were implemented into the KARATE-440 code system:
• more detailed parametrization of the few group constants (making the accurate calculations of the cores containing fuel relaxed for a longer time possible), corresponding renewal of the multigroup libraries and the parametrized few group constants,
• application of the more accurate calculation of the power distribution at the core periphery by using albedo matrices from Monte Carlo calculations,
• capabilities to handle the uncertainties of the basic nuclear data and the technological parameters.
The updated code has been verified by some standard calculations made for a VVER-440 core.

Reactor physics

316

2-D reflector modelling for VENUS-2 MOX Core Benchmark

Dušan Ćalić[1], Andrej Trkov[2]
[1]ZEL-EN razvojni center energetike, Hočevarjev trg 1, 8270 Krško, Slovenia
[2]International Atomic Energy Agency, Wagramerstr. 5, P.O.Box 100, A-1400 Vienna, Austria
dusan.calic@zel-en.si

The choice of the reflector model is important issue in full core calculations. In 2015 [1] new approach was developed where the existent WIMSD code for lattice cell calculations was replaced with Monte Carlo code Serpent 2 in order to have new reference full core calculation. However the Serpent-GNOMER simulation code uses simplified 1-D reflector model. In order to develop 2-D reflector model the VENUS-2 benchmark was proposed as a reference model. The main aim of this paper is to present the development of the 2-D reflector model based on VENUS-2 benchmark.

[1] Ćalić D., Trkov A., Monte Carlo Simulation on KRŠKO NPP. In: PHYSOR-2016. Sun Valley, Idaho, May 1-5, 2016.

317

3D Cartesian TRIGA reactor model quality assessment by radial power distribution

Vid Merljak[1], Andrej Trkov[2]
[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]International Atomic Energy Agency, Wagramerstr. 5, P.O.Box 100, A-1400 Vienna, Austria
vid.merljak@ijs.si

Numerical simulations provide useful insight into the behaviour of neutron population in a nuclear reactor. Indeed, defining the problem geometry is a crucial point where biases are introduced to a bigger or lesser extent. Recently, a Cartesian 3D geometrical model of the Jožef Stefan Institute's TRIGA Mark II research reactor has been developed for use with the GNOMER diffusion code. This model suffers from an inherent deficiency since the true reactor geometry is cylindrical and a Cartesian approximation had to be used to comply with the GNOMER's capabilities. Nevertheless, one can find good use of it. Experiments such as measurements of relative quantities (e.g. control rod reactivity worth) can easily be simulated. As a continuation of assessing the quality of the geometrical model in question, this paper presents comparison of radial power distribution as calculated by three different computer codes using various levels of model complexity: a cylindrical 2D model for TRIGLAV, a 3D Cartesian model for GNOMER and both Cartesian and detailed cylindrical 3D models for MCNP. Results are separated into two sections, namely determining the error due to geometry simplifications and the error due to approximations used in underlying theory (e.g. diffusion equation vs. Monte Carlo stochastic approach, the procedure of generating the nuclear cross-sections, etc.). It can be concluded that the GNOMER 3D Cartesian model is adequate for qualitative purposes. This is further confirmed by explaining most of the observed discrepancies. Combined with previous research, this paper represents a firm foundation for the model's practical use, particularly with respect of its strengths and weaknesses.

318

Simplified In Core Fuel Management Software for Education and Training

Erhan Şenlik, Mehmet Tombakoglu
Hacettepe University, Nuclear Engineering Department, 06800 Beytepe, Ankara, Turkey
mtombak@hacettepe.edu.tr

In this study, simplified in-core-fuel management software was developed to model one and two dimensional in core fuel management problems.
This study consists of six computer programs. These programs are based on one and two dimensional core loading pattern neutronic solvers and genetic algorithm optimization software. Neutronic parameters of Almaraz II Nuclear Power Plant data is utilized to perform power and burnup dependent full core calculations.
Simulation platform uses 1- and 2-D burnup dependent neutronic solver and they are coded in FORTRAN to acquire results quickly. To perform constraint and unconstrained optimization, genetic algorithm was developed and it is also coded in FORTRAN.
Remaining programs are graphical user interface programs which were coded in Python programming language. Calculation programs are; 1-DNodal, RPM-HUNEM and RPM-Genetic, graphical user interface programs are; Py1DNodal, PyRPM, PyRPM-Genetic.
1DNodal software is based on 1 dimensional core loading pattern. Py1DNodal software is a graphical user interface for

loading pre-chosen fuel types as an input to the 1DNodal software.
PyRPM code is used to specify fuel loading pattern, burn-up and power for RPM-HUNEM calculation software. PyRPM-Genetic graphical interface is designed for supplying number of fuel assemblies and required input parameters used in genetic algorithm software. The developed software has been tested using the benchmark data of Almaraz II Nuclear Power plant with different inputs and all are open for further development.

Reactor physics

319

Simulation of the Initial NPP Krško Cycles with CASL Core Simulator - VERA-CS

Andrew T. Godfrey[1], Fausto Franceschini[2], Mohamed Ouisloumen[2], Marjan Kromar[3]
[1]Oak Ridge National Laboratory, P.O.Box 2008, Oak Ridge, Tennessee 37831-6162, USA-Tennessee
[2]Westinghouse Electric Company LLC, 1000 Westinghouse Drive, Cranberry Twp 16066, USA-Pensylvania
[3]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
marjan.kromar@ijs.si

This paper describes the application of the Virtual Environment for Reactor Applications (VERA) core simulator (VERA-CS) under development by the Consortium for Advanced Simulation of Light Water Reactors (CASL), to the core physics analysis of the Krško NPP. VERA-CS aims at enabling whole-core fuel cycle depletion deterministic transport analysis with subchannel thermal-hydraulic coupling. It uses a three-dimensional (3-D) whole core transport code MPACT capable of generating sub-pin level power distributions. CTF, an improved version of the COBRA-TF subchannel code, is used for the calculation of the thermal-hydraulic parameters needed for the coupled calculation. This paper is focused on the application of VERA-CS to the analysis of the initial NPP Krško cycles. Obtained results are compared to the measurements performed during the plant operation. In addition results obtained from the Westinghouse PARAGON2/ANC9 system under development and IJS CORD-2 simulator are given also.

Reactor physics

322

Determination of the Computational Bias in Criticality Safety Validation of VVER-440/V213

Branislav Vrban[1], Jakub Lüley[1], Štefan Čerba[1], Filip Osuský[2]
[1]B&J NUCLEAR ltd., Alžbetin Dvor 145, 90042 Miloslavov, Slovakia
[2]Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Institute of Nuclear and Physical Engineering, Ilkovičova 3, 812 19 Bratislava 1, Slovakia
filip.osusky@stuba.sk

The key issue in any criticality safety problem is to estimate and to predict the deviation of calculation from reality. If the calculated value is not equal to its true value bias occurs. In criticality calculations the computational bias is the difference between the computed and the actual value of keff. The fundamental assumption is that the computational bias is mostly caused by errors in the cross-section data. In addition the use of random variables in the calculation introduces a non-random bias in the computed result as well. The American National Standards are utilized to predict and bound the computational bias of criticality calculations. These standards require the validation of the analytical methods and data used in nuclear criticality safety calculations to quantify the computational bias and its uncertainty. This paper presents a method for determining the computation bias and bias uncertainty for VVER-440/V213 reactor. For this analysis a SCALE

KENO 3D core model was developed by B&J NUCLEAR ltd company. This model is based on technical data and operational history of NPP Jaslovské Bohunice provided by the Slovenské elektrárne a.s. The operational conditions were defined for the end of campaign for which a fuel assembly-wise isotopic compositions in one sixth symmetry were calculated. The concentration of the boric acid was below 1 g per kg of water and the sixth group of control assembly was 18 cm below the upper position. Several calculation steps are used to address bias estimation method including sensitivity analysis, uncertainty analyses and cross section adjustment method. In addition the neutronic similarity of VVER-440/V213 core to several hundred critical benchmark experiments is evaluated by the use of three integral indices. The database of the benchmark experiment is based on the selection and processing procedure VALID provided by the Oak Ridge National Laboratory and specified in the ICSBEP Handbook. Systems with similar sensitivities to nuclear data uncertainties are expected to be computed to comparable accuracy so identification of similar integral experiments supports the accuracy of the determined computational bias. In cases where experimental benchmarks are available to validate specific nuclides, sensitivity and uncertainty analysis are used to project biases observed in the benchmarks to biases appropriate for the safety system. The results of all analyses performed are given and discussed in the paper.

Reactor physics

323

Recent development and examples of the use of the Windows interface environment XSUN-2016 for transport and sensitivity-uncertainty analysis

Ivan Aleksander Kodeli, Slavko Slavič
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
ivan.kodeli@ijs.si

In 2013 the first version of the Windows interface XSUN-2013 facilitating the deterministic radiation transport and cross-section sensitivity-uncertainty calculation was developed and submitted to OECD/NEA Data Bank Computer Code Collection. The package allows the preparation of input cards, rapid modification and execution of the complete chain of codes including TRANSX, ARTISN and SUSD3D in a user-friendly way. Recent updates of the code utility and several examples of its use will be presented, including cases such as:

- Transport, sensitivity and uncertainty analysis of the MYRRHA accelerator driven system (ADS), including both k_{eff} and β_{eff} parameters;

- Analysis of several benchmark experiments from the IRPhE and ICSBEP databases (SNEAK-7A & -7B, JEZEBEL, FLATTOP-Pu, etc.

The performance of the code system will be compared with those of other codes such as TSUNAMI, SERPENT and MCNP6.

Experimental and calculated data on criticality of uranium-zirconium hydride systems with 45% enriched uranium-235

Svyatoslav Sikorin, Siarhei Mandzik, Andrei Kuzmin, Tatsiana Hryharovich
The Joint Institute of Power and Nuclear Research-Sosny of the National Academy of Sciences of Belarus, PO BOX 119, 220109 Minsk, Belarus
sikorin@inbox.ru

The critical facilities “Rose”, “Edelweis”, “Liliya”, “Astra”, GFS, “Crystal” and “Giacint” of the Joint Institute for Power and Nuclear Research – Sosny of the National Academy of Science of Belarus have been used for over 45 years to generate and investigate more than a hundred uranium-containing critical assemblies with different material compositions, structures and targeted use, including uranium-water, uranium-alcohol, uranium-polyethylene and uranium-zirconium hydride multiplication systems and systems without moderators, with fuel rods and fuel assemblies with 10, 21, 36, 45, 75 and 90% enriched uranium-235, as well as with natural and depleted uranium. Also were researched uranium-water critical assemblies with rotating annular vortex core based of small diameter fuel particles with 90%-enriched uranium-235 and others multiplication systems. The uranium-zirconium hydride experiments were performed with 21, 36 and 45% enriched uranium-235. Beryllium, zirconium hydride and stainless steel were used in the reflector.
The paper presents criticality data produced at the critical facility “Crystal” for several uranium-zirconium hydride systems, representing non-uniform multiple zones heterogeneous uranium-zirconium hydride lattices comprising hexagonal fuel assemblies with cylindrical fuel rods, absorbing plates and rods, zirconium hydride and steel side and end reflectors. The critical assemblies represented the cores collected from three types of fuel assemblies with different structure, surrounded by assemblies and units of a side reflector. The core included channels for the regulating rods. The moderator – ZrH1.89. The fuel composition – UO2-Ni-Cr with 45 % uranium-235 enrichment. The absorber in plates – B with 85 % boron-10 enrichment. The absorber in rods – Eu2O3. The results of experiments on critical facilities with zirconium hydride have been analyzed by creating detailed calculation models. The analyses used the MCNP and MCU computer programs. The paper presents configurations of the studied uranium-zirconium hydride critical assemblies, as well as the experimental and calculation results.

402

Developing a New Neutron and Reactivity Monitoring System for Paks NPP

Sándor Kiss[1], Sándor Lipcsei[1], Gábor Házi[2], Zoltán Dezső[2], Tamás Parkó[3], István Pós[3], Miklós Ignits[3], László Hományi[4]

[1]Centre for Energy Research, Hungarian Academy of Sciences, Konkoly Thege M. út 29-33, H-1121, Hungary
[2]Centre for Energy Research, Hungarian Academy of Sciences, P.O.Box 49, H-1525 Budapest, Hungary
[3]MVM Paks Nuclear Power Plant Ltd., P.O. Box 71, H-7031 Paks, Hungary
[4]KFKI-Regtron Ltd, Hungary, Address, ZIP, Hungary
lipcsei.sandor@energia.mta.hu

The Reactivity Monitoring System and the Refuelling Neutron Monitoring System of Paks NPP are aged and need to be reconstructed. Since both systems are based on neutron flux measurements, the new system is to be served by the same detectors and measurement instrumentation. In order to provide data during refuelling, start-up and at full power, a full-range system is required, i.e. the detectors and the connected instrumentation should span the full range of neutron flux measurements from 0% to 100% of the reactor power. Additionally, the new system is required to operate continuously, to build a measurement archive, and to provide data for the Process Computer and the VERONA core monitoring system. In order to span the full neutron flux range, Photonis CFUL08 type ionisation chamber was chosen. The interface electronics will serve all three operation modes of the detector: impulse, Campbell (AC) and current (DC) modes. In order to obtain high reliability and dependability, the system will be built from independent and redundant components.

Reactor Operation

403

Systematic Approach to Training(SAT) for the design of Nuclear Power Plant(NPP) Decommissioning Training in South Korea

Jeong Keun Kwak

Korea Hydro & Nuclear Power Company, Ulsan, 45014, South Korea
bryan.kwak@khnp.co.kr

In 1979, the unavailability of Main Feedwater System(MFWS) in Three Mile Island(TMI) Nuclear Power Plant(NPP) Unit-2 happened. To make it worse, due to the malfunction of isolation valve control in Auxiliary Feedwater System(AFWS), the supply of cooling water to a Steam Generator(SG) was delayed approximately 8 minutes compared to a normal process in Abnormal Operating Procedure(AOP). In the long run, on account of deferred heat sink provision to a SG, the reactor core was melted partially. It was the first critical event in the US commercial NPP history. Therefore, after TMI accident, US Nuclear Regulatory Committee(NRC) suggested more than one hundred alternatives. Among them, one proposal was related to training area and it was Systematic Approach to Training(SAT) methodology. Hence, the goal of SAT is improvement of NPP stability through training point of view. Additionally, since the appearance of SAT in the NPP industry, it has been acquired the unwavering position in the NPP training field, so far.

Meanwhile, the significance of NPP decommissioning has been soared up in South Korea since the announcement of Kori NPP Unit-1 decommissioning determination. According to the proclaimed plan, Kori Unit-1 is scheduled to be decommissioned from June, 2017. Under this circumstance, nurturing sufficient number of proficient decommissioning engineers are one of the most urgent issue in South Korean NPP industry. Hence, to upgrade the efficiency and consistency

of training quality, SAT methodology can be the reliable solution for decommissioning training. For this reason, establishment of SAT based NPP decommissioning training will be a main considering factor in my paper.

Key words: Systematic Approach to Training, SAT, Nuclear Power Plant, NPP, Decommissioning, US Nuclear Regulatory Committee, US NRC, Three Mile ISland Unit-2, TMI Accident

Reactor Operation

404

DMReS, Digital Reacivity Meter of the new Generation

Slavko Slavič, Andrej Trkov, Bojan Žefran
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
slavko.slavic@ijs.si

A Digital Meter of Reactivity (DMR) was developed, which solves the point kinetics equations taking into account the source term. It is composed of a programmable picoameter, AD/DA converters, a PC computer and the accompanying software. From the start, the DMR proved to be superior to other similar devices, due to its ability to provide correct results in a very broad range of reactor operating conditions, including measurements in a deeply subcritical reactor. Special solutions was implemented in the DMR in order to cover broad range of cases.
The rod-insertion method makes full use of the special features available in the DMR. It is used for bank-worth measurements, during which a control bank is inserted into the core with the control rod drive mechanism at normal speed. No reactivity compensation is required; thus, the method is much faster than other available methods. The method has routinely been used during the start-up tests at the Krško NPP since 1990; it allowed to shorten the time required for the start-up tests from several days to only 12 hours. After the success in NEK, the rod-insertion method was adopted by others and is today used in several power plants around the world.
The DMR is used also for other measurements during start-up tests, e.g. for the isothermal temperature coefficient determination, perhaps the most important parameter that has to be satisfied at all times during a power reactor operation. The DMR device is indispensable for its determination. The temperature coefficient is defined as the change in reactivity with respect to the change of the reactor core temperature and must be negative throughout the cycle. During the measurement, the temperature of the coolant water is slowly lowered by approximately 2°C within half an hour and then raised again to the initial value. The reactivity is closely monitored. Since the absolute value of the coefficient is usually small at the beginning of each cycle, the accurate DMR has to be used for reactivity recording. The method enables very fast evaluation of data and the results are available immediately after the measurement.
The Windows version of the program, DMReS, was created with new graphical user interface (GUI). DMReS program uses a completely new way of reading data using DLL routines. Graphical representation of the results is completely new, making use of all the advantages of the Windows environment. Validation of the new DMReS code will be demonstrated in the paper.

405

Non-Destructive Testing of Reactor Pressure Vessel Nozzle

Petar Mateljak
INETEC-Institute for Nuclear Technology, Dolenica 28, 10250 Zagreb, Croatia
petar.mateljak@inetec.hr

The reactor pressure vessel (RPV) is an integral part of the reactor coolant pressure boundary. Ensuring safe operation of nuclear steam supply installation is a main and obligatory condition for the operation of all power units. One of the most important measures in fulfilling these requirements is periodical inspection of the condition of base metal, welded joints and RPV austenitic steel overlaying welding. As key part of reactor pressure vessel structural integrity, RPV nozzle sections in nuclear reactor pressure vessels are classed as critical components, requiring regular inspection to verify their integrity. Early detection of cracks is essential, however inspection costs are high. A plant shutdown costs operators an estimated €800,000 per day and a typical outage takes around 20 days to complete.
INETEC has recently developed new system for inspection of reactor vessel nozzle from the inside. Taking into account increasing requirements to the safety enhancement during plant operation, shortening of the inspection time, radiation exposure to examination personnel and cutting the total inspection costs, new system is designed as underwater automated robotic system with integrated NDT equipment. This paper describes the system's capabilities and features with focus on recent design evolutions.
Key words: reactor pressure vessel nozzle, non-destructive testing, automated inspection

Reactor Operation

406

A feasibility study on in-core fuel management via Quantum Particle Swarm optimization

Francesca Giacobbo[1], Gabriele Tavelli[2], Antonio Cammi[2], Marco Cauzzi[2]
[1]Politecnico di Milano, Department of energy, Via Ponzio 34/3, 20133 Milano, Italy
[2]Politecnico di Milano, Department of energy, Via La Masa 34, 20156 Milano, Italy
gabriele.tavelli@mail.polimi.it

Nowadays the increasing needs of optimizing the operations of fuel loading in a nuclear reactor core have been calling for efficient and reliable methods to determine suitable configurations of fuel assemblies able to maximize or minimize required neutronic or engineering features (i.e. multiplication factor keff, reactivity swing k, power peaking factor).
The current paper leads an investigation over already existing and widely employed optimization algorithms, aiming to target and further develop a promising method to tackle in-core fuel management optimization problems. Comparisons between algorithms were focalized on the fundamental goals of maximizing final result's accuracy and minimizing computational time required.
Given the intrinsic complexity of the issues under consideration, it looked reasonable to resort to global optimization methods to account for the possible existence of local optima. To this aim, Genetic Algorithms (GA) and Particle Swarm (PS) were selected, tested and eventually compared using an analytical multi-variable continuous test function with plenty of local optimum points.
The capabilities of both algorithms to not get stuck into local optima but to converge towards the global optimum were examined. Results obtained showed Particle Swarm to have better performance with respect to both accuracy and machine time. Therefore, the successive analyses were focused on particle swarm based algorithms such as Discrete Particle Swarm

(D-PS) (Kennedy and Eberhart, 1995) and its updated version, Discrete Quantum Particle Swarm (D-QPS) (Sun et al., 2004, 2005).
Given a number of fuel assemblies with different enrichments, the optimization of neutronic and engineering features is strictly related to the design of a precise core fuel spatial configuration.
Discrete Particle Swarm and a modified version of the original Discrete Quantum Particle Swarm algorithm, here proposed by the present authors, were tested with respect to their efficiencies and performances on a reactor physics case study.
The adopted case study regards an extremely simplified PWR core made of 5x5 fuel assemblies: 16 enriched with a content of U235 up to 3% (weight) and 9 enriched with a content of U235 up to 5% (weight). The goal was to optimize fuel assemblies disposition in order to obtain the highest value of the multiplication factor, keff. Given the extreme simplicity of this case study the global solution is a priori known. The calculations were performed coupling Particle Swarm based algorithms with Serpent Monte Carlo code. Obtained results showed a clear superiority of the Quantum Particle Swarm based algorithm proposed by the authors which speaks in favour about its application to more complex in-core fuel management optimization cases.

References

Kennedy, J., Eberhart, R., 1995. Particle swarm optimization. In: IEEE International Conference on Neural Networks (ICNN, 95), vol. 4. IEEE, Perth, Western Australia.

Sun, J., Feng, B., Xu, W.B., 2004. Particle swarm optimization with particles having quantum behavior. In: IEEE Proceedings of Congress on, Evolutionary Computation, pp. 325–331.

Sun, J., Feng, B., Xu, W.B., 2005. Adaptive parameter control for quantum behaved particle swarm optimization on individual level. In Proceedings of the 2005 IEEE International Conference on Systems, Man and Cybernetics, Piscataway, NJ, pp. 3049–3054.

Reactor Operation

407

Possibility of nuclear cogeneration development in the region of Paks

Török Szabina, Börcsök Endre, Talamon Attila
torok.szabina@energia.mta.hu

Almost half of GHG emission of world’s energy sectors is related to heat generation. The development of nuclear cogeneration offers a convenient possibility for emission reduction; however examination of economic constrains is essential. This study focused on heat demand of households in the vicinity of Paks NPP and compares economic and environmental aspects of domestic heating alternatives. In first part of our work we analyze competitiveness of nuclear cogeneration in district heating sector and in the second part we consider the optimal heating alternatives for different building typological groups taking into account economic and environmental aspects, distance from Paks NPP and heat demand density. We found that development of nuclear cogeneration is valuable above 17 €/tCO_2 price and with already existing district heating network. In districts with high heat demand density nuclear cogeneration based district heating can be competitive with stand-alone heaters if environmental externalities are also considered.

Thermal Hydraulics

501

Investigation of Thermal Turbulent Flow Characteristics of Wire-wrapped Fuel Pin Bundle of Sodium Cooled Fast Reactor in Lattice-Boltzmann Framework

Ali Tiftikci, Cemil Kocar
Hacettepe University, Nuclear Engineering Department, 06800 Beytepe, Ankara, Turkey
alitiftikci@hacettepe.edu.tr

In the presented study, thermal turbulent flow simulations of fuel pin bundles with helical spacer wires have been carried out. The lattice-Boltzmann method (LBM) is used for both fluid flow and heat transfer calculations. Hence, WALE and VLES turbulence models are implemented to the open-source LBM code and are coupled with the heat transfer modules. A 7-pin bundle geometry, flow and uniform heat flux conditions of Indian Prototype Fast Breeder Reactor (PFBR) are selected for the simulation purposes. The simulations are handled for hexagonal fuel rod bundle with two-helical-pitch-length geometry. The post-processed quantities such as velocity and temperature profiles and Reynolds stresses are compared for WALE and VLES turbulence models. Additionally, the Nusselt number and friction factor obtained from WALE and VLES are compared with the experimental correlations. The comparisons show that LBM simulations are in good agreement with the experimental data. Under the coarse (for WALE simulations) Dh/50 lattice resolution, VLES model gives relatively better results. In this study, it is also pointed out that LBM can be used for the complex fast breeder reactor coolant geometry and thermal turbulent flow conditions

Thermal Hydraulics

502

LOCA spectrum calculations for PWR by RELAP5 and TRACE

Andrej Prošek
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
andrej.prosek@ijs.si

The accident at the Fukushima Dai-ichi nuclear power plant in 2011 demonstrated that external events could cause loss of all safety systems. In the Europe stress tests were performed and the need was identified to further improve the safety of the existing operating reactors. Therefore the safety upgrade programs were started. The objective of this paper was to demonstrate that developed input model of two-loop pressurized water reactor (PWR) for TRACE thermal-hydraulic systems code has the capability for independent assessment of RELAP5 computer code calculations. For demonstration the response of PWR to loss-of-coolant accident (LOCA) was simulated. The break spectrum consists of 30.48 cm (12 inch), 20.32 cm (8 inch) and 15.24 cm (6 inch) equivalent diameter cold leg breaks. The initiating event was opening of the valve simulating the break. The reactor trip on (compensated) low pressurizer pressure (12.99 MPa) further caused the turbine trip. The safety injection (SI) signal was generated on the low-low pressurizer pressure signal at 12.27 MPa. On SI signal no active safety systems started (e.g. high pressure safety injection pumps and low pressure safety injection pumps and motor driven auxiliary feedwater pumps). Only passive components were assumed available, i.e. accumulators. All these LOCA scenarios with above assumptions lead to the core heatup. In this way the time available before significant heatup could be obtained.
For calculations the latest TRACE and RELAP5 computer codes were used: TRACE Version 5.0 Patch 4 using extension of Ransom and Trapp critical flow model (default) and RELAP5/MOD3.3 Patch 4 using Henry-Fauske critical flow model (default) and Ransom-Trapp critical flow model (Option 50).
The results showed that RELAP5 calculations using different break flow models are rather similar, therefore also other

parameters are similar. The accumulators discharge was faster in TRACE calculation than in RELAP5 calculations. Therefore the calculated TRACE break flow was also larger than RELAP5 calculated break flow during this period. It can be concluded that the break flow seems to be the largest contributor to the differences in the results between RELAP5 and TRACE.

Thermal Hydraulics

503

Prediction of low-pressure subcooled boiling with advanced interfacial area source term modelling

Ronak Thakrar[1], Simon P. Walker[2]
[1]Imperial College of Science, Technology and Medicine, Prince Consort Rd, London SW7 2BP, United Kingdom
[2]Department of Mechanical Engineering, Imperial College, Exhibition Road, London, SW7 2BX, United Kingdom
rkt08@imperial.ac.uk

Subcooled boiling flows are encountered frequently in the nuclear industry. There has been increased interest in the past decade to investigate low-pressure flows, primarily for advanced LWR concepts and research reactors. Mechanistic approaches to modelling nucleate boiling are in their infancy and remain a topic of intense research. Present-day CFD codes continue to rely heavily on conventional empirical modelling and have been applied widely towards the prediction of high-pressure flows in particular. In the current work, the Eulerian multiphase approach of the commercial CFD code STAR-CCM+ is applied to compute a vertically upward flow of water in a uniformly heated pipe near atmospheric pressure. Particular attention is placed on the prediction of the diameter of bubbles departing from the heated wall - a parameter that is referred to normally as a 'bubble departure diameter'. This parameter is an important modelling closure that affects a strong influence on the source terms for vapor generation and interfacial transport, and consequently on the generated void profiles. To this end, a more modern semi-mechanistic approach based on an intricate analysis of forces acting on the bubble is introduced into the established modelling framework. The predictive capability of this approach is compared and contrasted against conventional approaches. Prospects for improvement and suggestions for future investigation are outlined subsequently.

Thermal Hydraulics

504

Crack growth assessment in pipes under turbulent fluid mixing using an improved spectral loading approach and linear elastic fracture mechanics

Oriol Costa Garrido, Samir El Shawish, Leon Cizelj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
oriol.costa@ijs.si

The turbulent mixing of fluids with different temperatures inside of the pipes is a well-recognized source of thermal fatigue in the safety related piping of nuclear power plants. The fluid temperature fluctuations at the fluid-wall interface, caused by the turbulent mixing, induce temperature fluctuations in the surrounding pipe walls. Rather fast temperature fluctuations at the pipe surface induce fluctuations of localized thermal strains which are constrained by the adjacent material at different temperature. In this way, the fluid temperature fluctuations induce stress fluctuations in the pipe, which may lead, in some circumstances, to fatigue and subsequent leakage or even loss of structural integrity. This phenomenon is also known as thermal stripping.
This paper estimates the probability of surface crack growth through the pipe wall under turbulent fluid mixing conditions

using a damage tolerant approach. It is usually believed that high frequency oscillations of fluid temperatures during turbulent mixing may be responsible for crack arrest. In these conditions, large stress gradients in thickness direction are the cause for the reduction of the stress intensity factor as the crack grows. However, these observations are typically derived from crack growth analyses assuming that fluid or pipe surface temperatures follow single frequency sinusoidals.
The growth rates of stipulated surface cracks are studied for diverse variations of fluid temperatures, generated with an improved spectral loading approach recently developed by Jožef Stefan Institute. The analyses use a rather simple and linear one-dimensional model of the pipe with numerically resolved time-dependent temperatures and analytical expressions for the linear elastic wall thermal stresses varying only in the radial direction. The linear elastic fracture mechanics theory is then employed to compute the time-dependent stress intensity factors of the crack following the method of weight functions for general stress profiles. The uncertainties of crack growth, which arise from the use of comparatively short fluid temperature histories to the expected fatigue life time of months or years, are moreover evaluated for diverse time lengths of the fluid temperature histories using the rainflow counting method and the Paris law. The likelihood of crack arrest is finally assessed for the diverse variations of the fluid temperatures.

Thermal Hydraulics

505

Simulation of the Experiment PKL III H2.1 with the TRACE5 Code

Jara Turégano Lara, Maria Lorduy, Sergio Gallardo, Gumersindo Verdú
Department of Chemical and Nuclear Engineering, Polytechnic University of Valencia, Camí de Vera sn, 46022 Valencia, Spain
sergalbe@iqn.upv.es

In the nuclear safety it is especially important to know the thermal hydraulic phenomena which take place during an accident in a nuclear power plant. However, it is impossible to perform the experiments in a real scale in order to obtain the data. For this reason, there are facilities that replicate some nuclear power plants scaled. Such is the case of the PKL facility of AREVA placed in Erlangen, Germany, which models the most important components of the primary and secondary side.
The purpose of this work is the analysis of the results obtained with a PKL model developed with TRACE5 Patch 4 and its comparison with the experimental results. PKL-III H2.1 experiment is based on a Station Black Out (SBO) accidental scenario with secondary and primary side depressurization. This experiment consists of 4 stages (conditioning, A, B and C phases). The main thermal hydraulic variables (pressure, flow, temperatures, CET and PCT) have been studied.
Two models of PKL have been used (1-D vs 3-D vessel components) in order to compare the behavior of the coolant from both of them to the real vessel. The obtained results show that the 3D model is able to simulate the main thermal hydraulic phenomena occurring during the conditioning phase and the accidental phase, among them the core uncovery and the rise of the temperatures. The results with both models have been compared with the experimental results. It can be concluded that, for this particular case, the 3D-vessel component allows a proper representation of the vessel collapsed water levels as well as the Core Exit Temperature (CET) and the Peak Cladding Temperature (PCT). Moreover, the effect on the steam generators and vessel emptying have been studied when a finer nodalization in the U-tubes and the vessel is applied. As a general conclusion, TRACE5 reproduces the main phenomena observed in this test successfully.

507

On the discontinuity of the dissipation rate associated with the temperature variance at the fluid-solid interface for cases with conjugate heat transfer

Cedric Flageul[1], Sofiane Benhamadouche[2], Eric Lamballais[3], Dominique Laurence[4], Iztok Tiselj[1]

[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]EDF R&D, Fluid Mechanics, Energy and Environment Dept., 6 Quai Wattier, 78401 Chatou, France
[3]Institute PPRIME, Department of Fluid Flow, Heat Transfer and Combustion, Université de Poitiers, CNRS, ENSMA, Téléport 2 – Bd. Marie et Pierre Curie, B.P. 30179, 86962 Futuroscope Chasseneuil Cedex, France
[4]The University of Manchester, Materials Performance Centre, School of Materials, PO Box 88, M60 1QD Manchester, United Kingdom
cedric.flageul@ijs.si

Conjugate heat transfer describes the thermal coupling between a fluid and a solid. It is of prime importance in industrial applications where fluctuating thermal stresses are a concern, e.g. in case of a severe emergency cooling (PTS) or long-term ageing of materials (T junctions). For such complex applications, investigations often rely on experiments, high Reynolds RANS or wall-modelled LES. However, experimental data on conjugate heat transfer are scarce as walls in lab rigs are often made of plexiglas and the transported scalar studied is often a dye. The development of RANS models for conjugate heat transfer is relatively recent (Craft et al., Journal of turbulence, Vol. 11, 2010). In this paper, we establish that the dissipation rate associated with the temperature variance is discontinuous at the fluid-solid interface, in case of conjugate heat transfer. There is currently no RANS model for conjugate heat transfer that takes into account this discontinuity.

Thermal Hydraulics

508

Thermal-hydraulic Analysis Code for Plate-type Fuel Nuclear Reactors

Duvan Alejandro Castellanos Gonzalez, Pedro Carajilescov, Jose Maiorino

Universidade Federal do ABC - PROGRAMA DE PÓS-GRADUAÇAO EM ENERGIA, Av. dos Estados, 5001. Bairro Bangu. Santo André - SP , 09210-580, Brazil
duvan.castellanos@ufabc.edu.br

The use of plate-type fuel assembly, in nuclear reactors, are mostly associated to researched reactors and naval propulsion reactors (aircraft carriers and submarines), bringing immediate benefits in security and thermal-hydraulic performance of the reactor. Computational codes are used to calculating the thermal-hydraulic core behavior. This project presents the development of thermal-hydraulic code for reactors with plate-type fuel elements, written in FORTRAN. According to geometric input data, operational and boundary conditions, the code involves the analysis of steady state flow and power regime, solving the conservation equations for mass, momentum and energy. Furthermore, it performs the calculation of minimum DNBR, based on the analysis of critical channel. The code has maximized the radial mesh with the use of the chain or cascade method for two stages: in the first stage, the core is subdivided in sub channels with size equivalent to a fuel assembly and the second stage, the hot fuel assembly is subdivided in sub channels with size equivalent to the one channel that comprise. For the program validation, it was considered the research reactor CARR (China Advance Research Reactor), and the LABGENE reactor (Brazilian reactor of naval propulsion). The code yields detailed information of reactor core as the change of the static pressure in the channel, flux distribution, variation of coolant temperature and coolant

velocities, quality and local flux heat in the critical channel. The analysis showed good agreement compared to the results obtained for CARR reactor and for a typical reactor power PWR.

Thermal Hydraulics

509

The influence of imposed gas velocity profile on wave dynamics in the simulation of vertical air-water churn flow

Matej Tekavčič, Boštjan Končar, Ivo Kljenak
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
matej.tekavcic@ijs.si

A three-dimensional transient simulation of isothermal churn flow of air and water in 19 mm internal diameter vertical pipe was performed. The churn flow regime in vertical pipes can be viewed as a transitional regime between slug flow and annular flow and is often related to the onset of the flooding phenomena, which is of particular interest for safety analyses of the loss-of-coolant accident in light water nuclear reactors.

Single-fluid interface capturing approach based on the volume of fluid method with interface compression was used to model the gas-liquid interactions. A short section of a vertical pipe with perforated wall was simulated representing the region near a typical liquid inlet section in experiments, where large waves of liquid travelling upwards can typically be observed in the churn flow regime. One of our previous attempts to model such flooding type liquid waves with computational fluid dynamics approach showed a systematic over-prediction of wave frequencies compared to the values reported from experiments available in the open literature.

The influence of different gas velocity profiles on the wave dynamics in the pipe is investigated in the present paper. Three different types of gas velocity profiles imposed at the bottom of the vertical pipe computational domain were chosen to represent a model for gas flow predicted by k-omega SST turbulent model and two extreme deviations from it. The flow conditions – the values for mass flow of gas and liquid - are taken from experiments from the literature with gas Reynolds number between 7000 and 10000. The effect of the gas velocity profile on the calculated frequency of waves is presented. Results for time-averaged axial and radial profiles of pressure, velocity, and liquid volume fraction are presented.

Thermal Hydraulics

510

Analysis of the MSIV Closure Transient Simulation in APROS

Tadeja Polach[1], Ivica Bašić[2], Luka Štrubelj[3]
[1]ZEL-EN razvojni center energetike, Hočevarjev trg 1, 8270 Krško, Slovenia
[2]APoSS d.o.o., Repovec 23b, 49210 Zabok, Croatia
[3]GEN energija d.o.o., Vrbina 17, 8270 Krško, Slovenia
tadeja.polach@zel-en.si

The Slovenian Krško Nuclear Power Plant (NEK) model was built in using APROS - Advanced PROcess Simulation environment. The basis for this model was the RELAP5/MOD3.3 Engineering Handbook, the model was updated to the 26th cycle and also includes the upflow conversion modification.
A detailed model nodalisation was created for each system and every system was separately validated. The current model

covers the primary circuit with the core kinetics model, the secondary circuit and their control systems. The steady state model already having been validated the plan is to validate the model for some transients and design basis accidents. In this article the plant behaviour after the Main Steam Isolation Valve (MSIV) closure. Two scenarios of the closure are performed. In the first both MSIVs are closed at the full power operation and in the second only one MSIV is closed, again at full power operation.

Upon initiation of the of the MSIV closure the control system signal actuations and their times were followed and the responses of different affected systems were being observed. All those recorded values were then compared with the identical transient performed on the similar NEK model with the RELAP5/MOD3.3 system code. This procedure allowed to bring the current APROS NEK model one step forward towards being assured to have accurate calculations.

Thermal Hydraulics

511

Spectral Element Direct Numerical Simulation of Sodium Flow over a Backward Facing Step

Jure Oder, Jernej Urankar, Iztok Tiselj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
jure.oder@ijs.si

In this paper we present the direct numerical simulations of a turbulent flow of a liquid metal past the backward-facing step (BFS) with finite dimensions. The BFS geometry can be visualised as a channel, where one of the walls has a shape of a step. The flow is flowing from the narrower part to the wider part. The simulations are performed in three dimensions.

For the inflow boundary condition over the BFS, a fully developed turbulent velocity field is used. To obtain this fully developed turbulent inflow, a separate domain is constructed. It has a geometry that corresponds to the geometry of the channel before the step, the narrow part. The boundary conditions within this subdomain are set to be periodic in the direction of general flow. A velocity field in a plane from this subdomain is then used as an inflow boundary condition for the flow over the BFS. The flow in the subdomain is calculated simultaneously with the simulation of the flow over the step.

Simulations are performed with the NEK5000 code. The most notable feature of this code is the use of spectral elements to solve for velocity, temperature and any other passive scalar. It is an open source code developed by the Argonne National Laboratory.

Spectral element method is a hybrid method between finite element method and a collocation spectral method. The method divides the computational domain into finite elements, within which a spectral method is used to solve for variables. This method allows for the use of spectral method in irregularly shaped geometries and to perform direct numerical simulations in such geometries.

The main purpose of this work is to test the numerical set-up to later perform calculations with temperature field as a passive scalar. Dimensional walls with internal heating will be added to simulate the heat production in the walls.

This work is part of work that is performed within the SESAME project of Horizon2020 research programme and is a continuation of research at our department.

512

Development of Turbulent Mixing Layer in Horizontal Confined Two-Component Flow

Rok Krpan, Boštjan Končar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
rok.krpan@amis.net

Interaction of two fluid liquid streams with different densities and temperatures is of particular interest in nuclear industry. Notable examples are turbulent mixing in junctions of primary coolant piping or boron mixing in reactor core of the pressurized water reactor. The turbulent flow phenomena in the wake mixing zone, in which two streams interact, develop differently depending on the physical properties of the two liquids involved. The purpose of this work is to predict the development of turbulent mixing layer due to mixing of two water streams with slightly different densities in a horizontal square duct. The mixing of such flows can be modelled as the flow of two components, where the concentration of one component in the mixing zone can be described as a passive scalar. Velocity field remains common over the entire computational domain and is affected by density difference due to concentration. Different CFD codes and turbulence models will be used for simulations and the results will be compared with experimental data. The main goal of the study is to demonstrate the capabilities of different codes and modelling approaches to predict the differences in turbulent flow phenomena in the wake mixing zone between the single liquid case and two-component liquid case. Computational results obtained with OpenFOAM and ANSYS FLUENT code will be compared with GEMIX experimental data, obtained from Paul Scherrer Institute in Switzerland.

Thermal Hydraulics

513

Experiments on bubbly to slug flow transition in a vertical cylindrical tube

Matic Kunšek[1], Daisuke Ito[2], Yasushi Saito[2]
[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]Kyoto University Research Reactor Institute, Osaka, Japan
matic.kunsek@ijs.si

Two-phase fluid flows can be found in numerous industrial installations, such as nuclear power plants, chemical, biological and biomedical reactors, within different industrial fields. A special case of two-phase flow, which is excellently suited for experimental work, is air-water two-phase flow. Both fluids are easily accessible and, with their use, adiabatic system of steam-water is easily simulated. This system enables one of the key processes in nuclear and other power plants as well. Steam-water two-phase flow can be found in the cores at working conditions in boiling water reactors, in all steam generators and in the cores of pressurized water reactors during some hypothetical accidents. Because of that, the knowledge of behavior of those systems is very important. Although two-phase flows can be found all around us and that research field exists for decades, a lot of questions still remain unanswered because of complexity of interactions between phases.

In the proposed paper, measurements of void fraction in bubbly and bubbly to slug flow transition regimes are presented. First, the description of wire mesh sensor construction is shown. Then, the measurements of two-phase air-water flow at 20 different flow conditions (liquid and gas flow rate) are described. In the end, the processing of data is explained and the results are presented, analyzed and the bubbly to slug flow transition is identified.

The work for the proposed paper was done at the Heat transfer laboratory of Kyoto University Research Reactor Institute

(KURRI) in Osaka prefecture in Japan.

Thermal Hydraulics

514

Simulation of bubbly to slug flow transition in a vertical cylindrical tube with OpenFOAM computer code

Matic Kunšek, Ivo Kljenak, Leon Cizelj
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
matic.kunsek@ijs.si

Two-phase fluid flows can be found in numerous industrial installations within different industrial fields. A special case of two-phase flow is air-water two-phase flow. With its use, it is easy to simulate adiabatic systems of steam-water which are part of the most important systems in nuclear power plants. Steam-water two-phase flow can be found in the cores at working conditions in boiling water reactors, in all steam generators and in the cores of pressurized water reactors during some hypothetical accidents. The use of air-water is excellently suited for comparison with experimental work which is necessary to validate the results of simulations. Because of the importance, a lot of studies have been done in last decades. However, a lot of questions still remain unanswered because of complexity of interactions between phases.
In the proposed paper, simulations of bubbly flow and bubbly to slug flow transition regimes are presented. First, the descriptions of used numerical mesh, equations and models are provided. The two-fluid model of two-phase flow, implemented in the open-source OpenFOAM computer code, was used. Then, the simulations of two-phase air-water flow with the OpenFOAM code are presented. In the end, the processing of data is explained and the results (void fraction profiles) are compared with measurements.
The work for the proposed paper was done at the Jozef Stefan Institute (JSI), Slovenia. The measurements used for comparison were made at Heat transfer laboratory of Kyoto University Research Reactor Institute (KURRI) in Osaka prefecture in Japan.

Thermal Hydraulics

515

Evaluation of Non-condensable Gas Effect on the Operation of Emergency Core Cooling System during LBLOCA

Seunghun Yoo, Kwang-Won Seul, Young-Seok Bang
Korea Institute of Nuclear Safety, 34 Gwahak-ro, Yuseong-gu, Daejeon 305-338, South Korea
k720ysh@kins.re.kr

The gas accumulation in the piping of diverse fluidic systems has occurred since first commercial operation of nuclear power plants. If the gas is accumulated in the Engineered Safety Features (ESF) such as Emergency Core Cooling System (ECCS), the gas can increase the potential to damage the pipe and components and it may cause the condition that the ESFs is inoperable during the Design Basis Accident. On the basis of the pump which is an important active component in the ESFs, if the gas is accumulated in the pump suction, it can induce the pump cavitation or influence the other pumps sharing common suction pipe. If the gas is accumulated in the pump discharge pipe, it can cause the water hammering and damage the pipe and its associated systems. Despite of the significance of the gas accumulation consequences, the gas transport and its effect on the Design Basis Accident were not comprehensively considered in the current Final Safety Analysis Report. Therefore it is necessary to understand and clarify the consequences of the gas accumulation in the ESFs and during the

Design Basis Accident.
In this study, the gas accumulation effect on the operation of the ECCS during Large Break Loss of Coolant Accident (LBLOCA) was evaluated as dividing two analyses: a) the effect on High and Low Pressure Safety Injection Pumps (HPSIP and LPSIP) with the condition that the gas is located in the pump suction, b) the gas transport behavior in the ECCS during LBLOCA. Shin-Kori unit 1 and 2 was selected as a target plant for this study. RELAP5/Mod 3.3 Patch 4 was used to model HPSIP, LPSIP and ECCS. For the first analysis, the single phase homologous curve and the two phase head multipliers for the pumps were newly modified to reflect the characteristics of Shin-Kori's SIPs. The hypothetical non-condensable gas was injected into the pump suctions and the changes of the pump suction void fraction, suction flow regime, pump velocity, head and flow rate were evaluated to quantify the degradation of the pumps. And the flow condition in the pipe was evaluated as calculating the Froude Number. For the second analysis, the precise piping system for the pump upstream which was covered from Refueling Water Tank to each pump suction was modeled as reflecting minor losses caused by geometrical change of pipe, valve or orifice, etc. As assuming that the gas can be randomly accumulated in the ECCS, the gas transport trajectory for the randomly accumulated points and finally transported locations in the ECCS were identified. On the basis of the analysis, the gas transport behavior for the hypothetical amount of gas which located in the final accumulation points wad evaluated as simulating the LBLOCA condition that modeled by the cold leg's pressure and temperature changes using the time dependent volume. Moreover, the flow condition for the ECCS during LBLOCA was analyzed by calculating the Froude Numbers for diverse locations.
As a result, we have quantified the degradation of the Shin-Kori's SIPs due to non-condensable gas. When the gas is randomly accumulated at the diverse points of the ECCS, the finally transported locations were identified. And the gas transport behavior in the ECCS during the LBLOCA was analyzed and vulnerable points were identified. The flow conditions for the whole ECCS locations, which may help to judge whether the gas is transported or not, was evaluated as the Froude Number.

Thermal Hydraulics

516

Transient Response of Typical VVER Steam Generator Based on RELAP and Simplified Models

Hüseyin Ayhan, Cemal Niyazi Sökmen

Hacettepe University, Nuclear Engineering Department, 06800 Beytepe, Ankara, Turkey
huseyinayhan@hacettepe.edu.tr

The steam generator is a highly complex and nonlinear system and has time varying dynamics. Steam Generator (SG) parameters vary with operating conditions like feed water flow rate or temperature. The main goal of SG control system is to maintain the SG water level at a desired value by regulating the feed-water flow rate. In the present study, transient response of horizontal SG is investigated using RELAP5 Mod3.4 system code and simplified controller model. Responses were compared and performance of water level controller model was studied.

Controller model (E. Irving, 1980) that is used in this study is usually adopted for U-tube SG. In this study this model is applied for horizontal SG and its performance is tested using comparisons with RELAP simulations. Comparisons are made for both low and high operating power. Simulations and model responses show that the level controller can predict transient response of horizontal steam generator water level through different operating power.

517

Modeling of NEK Steam Line Break analysis in computer code Apros 6

Jure Jazbinšek[1], Luka Štrubelj[2], Klemen Debelak[2], Ivica Bašić[3]
[1]ZEL-EN razvojni center energetike, Hočevarjev trg 1, 8270 Krško, Slovenia
[2]GEN energija d.o.o., Vrbina 17, 8270 Krško, Slovenia
[3]APoSS d.o.o., Repovec 23b, 49210 Zabok, Croatia
jure.jazbinsek@zel-en.si

Model of Nuclear power plant Krško (NEK) developed in computer code Apros 6 was upgraded with reactor containment building and compared to Krško NPP RELAP and GOTHIC code models developed by FER.
Best estimate Apros 6 analysis of double-ended Main Steam Line Break (MSLB) transient was simulated. MSLB transient between Steam Generator (SG) outlet and Main Steam Isolation Valve (MSIV), so the blowdown of affected SG could not be prevented, was assumed.
When the break occurs, the rapid steam flow to containment building occurs from affected SG, causing rapid cooling and pressure drop of Reactor Coolant System (RCS). Containment absolute pressure, temperature and liquid void fraction in Apros 6 simulation will be compared to results of GOTHIC code. The discharged liquid and gas volume from MSLB event and corresponding main control room variables in Apros 6 simulation will be compared to RELAP5 results.

518

Assessment of Condensation Heat Transfer Models of MARS-KS and TRACE Codes Using PASCAL Test

Kyung Won Lee[1], Ae-Ju Cheong[2], Andong Shin[2]
[1]Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon, 34142, South Korea
[2]Korea Institute of Nuclear Safety, 34 Gwahak-ro, Yuseong-gu, Daejeon 305-338, South Korea
leekw@kins.re.kr

The advanced power reactor plus (APR+) is a GEN-III+ nuclear power plant, the standard design of which is currently being developed in Korea. The passive auxiliary feedwater system (PAFS) is one of the advanced safety features adopted in the APR+ and is design to replace the conventional active auxiliary feedwater system. During the plant transient, PAFS cools down the secondary side of steam generator, and eventually remove the decay heat of the reactor core by condensing steam in nearly-horizontal U-shaped tubes (passive condensation heat exchanger, PCHX) submerged inside the passive condensation cooling tank (PCCT).

In order to validate the operational performance of the PAFS, Korea Atomic Energy Research Institute (KAERI) has performed the experimental investigation using the PASCAL (PAFS Condensing heat removal Assessment Loop) facility. The PASCAL simulates a single U-shaped tube with the volumetric scaling ratio of 1/240. The dimension and material of the tube are the same as the prototype. The inner and outer diameter of the PCHX are 44.8 mm and 50.8 mm, respectively. The tube length is 8.4 m. The width and depth of the PCCT are 6.7 m and 0.112 m, respectively. The height of the PCCT is 11.484 m.

In this study, we assess the applicability of the condensation heat transfer models of MARS-KS and TRACE codes to the nearly-horizontal condensing tube using the PASCAL SS-540-P1 test. In the MARS-KS input model, the PCHX, is modeled

using the PIPE component with the 28 cells. The steam-supply line and the condensate-return line are modeled using the time dependent volumes. The PCCT is modeled with the multi-D component. The HTSTR component is used to model the heat transfer between the PCHX and the PCCT. The TRACE input model has the same nodalization as the MARS-KS input model. The PCCT is modeled with the 3-D vessel component.

The calculation results of heat flux, steam and condensate temperatures are compared with the experimental data. The results show that MARS-KS slightly under-predict the heat fluxes. However, TRACE over-predicts the heat fluxes at the tube entrance region and under-predicts the heat fluxes at the tube exit region. When compared to MARS-KS results, TRACE provides more reasonable condensate temperatures.

Thermal Hydraulics

519

The Effect of Tube Arrangement and Turbulence Models for Steady Flow Past Tube Bundles

Ali Tiftikci, Cemil Kocar
Hacettepe University, Nuclear Engineering Department, 06800 Beytepe, Ankara, Turkey
alitiftikci@hacettepe.edu.tr

In present work, three-dimensional flow in tube bundles for different tube configurations is studied. In-line and staggered type bundle arrangements are used. The tube configurations are 3.6x1.6 and 3.6x2.1 for staggered arrays and 3.6x2.1 for in-line arrays. The lattice-Boltzmann method is used for numerical calculations. The validation processes are made by using the experimental data available in ERCOFTAC Database (Case 80). The turbulence models LES (Smagorinsky-Lilly) and VLES (k-omega) are selected for comparison. Also, mesh sensitivity analyses are made. Post processed quantities such as axial and transverse mean velocity and corresponding rms (root mean square) velocity profiles at different locations are compared with the experimental data. The simulation results of LES and VLES are in good agreement with the experiment for different tube bundle arrangements.

Thermal Hydraulics

521

Simulation of a station blackout transient using TRACE5. Application to ATLAS facility.

Maria Lorduy[1], Jara Turégano Lara[2], Sergio Gallardo[1], Gumersindo Verdú[1]
[1]Universidad Politecnica de Valencia, Departamento de Ingeniería Química y Nuclear, Camino de Vera s/n, 46022 Valencia, Spain
[2]Department of Chemical and Nuclear Engineering, Polytechnic University of Valencia, Camí de Vera sn, 46022 Valencia, Spain
sergalbe@iqn.upv.es

The purpose of this work is to test the capability of the TRACE5 code in the simulation of a Station Black Out (SBO) transient with delayed asymmetric secondary cooling in the frame of the OECD-ATLAS project. In this proposal, test A1.1 is analysed. A TRACE5 model of the ATLAS facility has been developed in order to simulate both steady state conditions and the SBO transient. This facility is designed to simulate transients and accidents in APR1400 reactors, operating under the same prototypic pressure and temperature of its reference plant. Due to the especial characteristics of this transient regarding

to the possible asymmetries, it is necessary to study the 3D behaviour of coolant in the vessel. A 3D-vessel component has been used to simulate the real vessel of the facility. In this work, a detailed analysis of some components of the model is performed, especially of the pressurizer and the steam generators relief valves. Furthermore, main steam lines and the line to the refuelling water tank is analysed to best estimate the water discharge performance.
The test simulates a prolonged station black out with delayed asymmetric secondary cooling through the supply of auxiliary feedwater only in one steam generator. The main goal of this test is to analyze the primary cool-down performance by delayed asymmetric secondary cooling as an accident mitigation measure. The transient comprises two phases: the first phase simulates the station blackout transient without auxiliary feedwater, and the second phase simulates the activation of auxiliary feedwater only in one steam generator.
The transient is analysed by means of the main thermal hydraulic variables: primary and secondary pressures, mass flow rates, discharged coolant inventory through the pressurizer and steam generators relief valves, collapsed water level in the vessel and in the pressurizer, fluid temperatures in hot and cold legs, etc.
The results show that the TRACE5 code is able to successfully reproduce the main physical phenomena observed during the experiment.

Thermal Hydraulics

522

Prediction of Wall Condensation in the Presence of Non-Condensable Gases through Various Thermal-Hydraulic Codes

Erol Bicer, Yeon-Joon Choo, Seong-Su Jeon, Seung-Sin Kim, Yong-Hwy Kim, Soon-Joon Hong
Future and Challenge Technology Co., Ltd. (FNC Tech.) , 46 Tapsil-ro, Giheung-gu, Yongin-si, Gyeonggi-do, 17084, South Korea
ebicer@fnctech.com

Vapor film condensation is an important topic in various Light Water Reactor (LWR) nuclear safety applications. In most cases, condensation takes place in the presence of non-condensable gases, such as the condensation along containment walls in the presence of air following a Loss of Coolant Accident (LOCA). Recently, various Passive Containment Cooling Systems (PCCS) are under design and are expected to operate with high non-condensable gas content. Thus to determine whether the current Lumped Parameter (LP), and/or Multi-Dimensional simulation codes can accurately simulate the condensation phenomenon will be investigated. For this purpose, four of COPAIN facility tests (both forced and natural convection) were simulated through MARS-KS, GOTHIC, CUPID, and CFX codes. Early results indicate that the one dimensional LP codes could not provide accurate results (usually underestimation) when the flow dynamics is three dimensional. One main reason of the inaccuracy is due to the empirical correlations that are used in the condensation models. Furthermore, the velocity vector maps and temperature contour maps are provided for further investigation on heat flux along the wall. Experimental data from COPAIN facility are compared with the simulation results to evaluate the applicability of wall condensation model of each code.

523

UHS Cooling Pond Evaluation using NUREG-0693 Methodology

Davor Grgić[1], Nikola Čavlina[2], Tomislav Fancev[1]
[1]University of Zagreb, Faculty of Electrical Engineering and Computing , Unska 3, 10000 Zagreb, Croatia
[2]Fakultet elektrotehnike i računarstva Zagreb, Unska 3, 10000 ZAGREB, Croatia
davor.grgic@fer.hr

The main objective of a Ultimate Heat Sink (UHS) is to provide cooling water for nuclear power plant safety related systems. It dissipates residual heat after reactor shutdown and after an accident through cooling components of the Essential Service Water (ESW) System and the Component Cooling Water (CC) system. All accident analyses described in SAR Chapter 15 are performed taking into account some boundary conditions directly or indirectly based on UHS temperature. From licensing (US NRC) point of view UHS role is covered by three General Design Criteria (GDC) of Appendix A to 10 CFR Part 50, GDC 2 "Design bases for protection against natural phenomena", GDC 5 "Sharing of structures, systems and components", and GDC 44 "Cooling water". Main UHS licensing requirements are formulated in Regulatory Guide (RG) 1.27 "Ultimate Heat Sink for Nuclear Power Plants", and acceptable supporting analysis, at least for the case when cooling pond is used as UHS, is described in NUREG-0693, "Analysis of Ultimate Heat Sink Cooling Ponds".
NPP Krsko is using Sava river as one ultimate heat sink with dual role. In normal situation the river flow is directly providing cooling water for ESW heat exchanger (the requirements are in the form of minimum river flow rate (guaranteed NPSH for ESW pumps) and maximum allowed water temperature), and in highly unlikely situation that river flow is stopped, remaining water is forming cooling pond upstream of the river dam (the requirements are in the form of minimum initial required water volume and its temperature during mentioned interval).
In this paper the analysis using NUREG-0693 methodology was performed to check behavior of NPP Krsko cooling pond temperature and volume during time interval of 30 days after DBA LOCA coincident with loss of Sava flow. NUREG-0693 uses a simple mathematical model of a cooling pond to scan weather data to determine the period of the time for which the most adverse pond temperature or rate of evaporation would occur. Once the most adverse conditions are available, the peak pond temperature is determined for given heat load coming from the plant.

524

Estimation of SFDS Cask Heat-up after Blockage of Ventilation Openings

Davor Grgić, Siniša Šadek, Vesna Benčik
University of Zagreb, Faculty of Electrical Engineering and Computing , Unska 3, 10000 Zagreb, Croatia
davor.grgic@fer.hr

Spent Fuel Dry Storage (SFDS) is becoming, especially after Fukushima accident, popular alternative to spent fuel pool storage of spent fuel elements. Usual requirements for SFDS casks are structural robustness in all conditions, guaranteed subcriticality of the content, low contact dose rate and adequate passive cooling of the fuel. Two basic designs of the SFDS casks exist, one using metal and other using concrete casks. One of the usual assumptions in safety analysis of the concrete cask design is loss of natural circulation between cask's insert and body of the cask due to inlet and outlet openings blockage. It is required to demonstrate how long it takes till reaching limiting fuel cladding temperature in adverse conditions. The simple calculation model for GOTHIC code was developed based on publically available data (Safety Analysis Report) for HOLTEC vertical concrete cask system. For given heat loading steady state temperature distribution is calculated as well as subsequent heat-up during extended period of time (7 days) after closing top and bottom ventilation holes. In

the specific case it took 5 days to reach the targeted temperature of 570 oC (with more conservative assumptions it could be less), but taking into account that only approximate data were available for geometry of the cask, the model is mainly used for estimation of heat-up trends and for quantification of different heat transfer mechanisms relative importance.

601

Microstructural evaluation of creep behavior in hydrided E110 cladding

Hygreeva Kiran Namburi
Research centre Rez, Hlavni 130, 250 68 Husinec-Řež, Czech Republic
hygreeva.namburi@cvrez.cz

Creep is considered as one of the dominant damage mechanisms in zirconium based fuel cladding's during reactor operation and spent fuel in dry storage. This paper emphases on the microscopic examination results from the E110 cladding in its virgin state and after high temperature creep test at dry storage conditions. Test specimen was oxidized in an autoclave at a pressure of 10.7 MPa and temperature of 425 °C (150 ppm by weight H). Creep test was performed at UJP, Praha in a horizontal furnace under the conditions: internal pressure - 4 MPa, temperature -530 °C, exposure time -30 hours.

TEM observations were made on creep tested specimen to reveal interaction of dislocations, secondary precipitate particles and hydrides with grain boundaries owing to different deformed zones. The material is characterized by polyhedral grains of α - Zr phase with hexagonal lattice and lattice parameters a = b = 3.136 Å, c = 5.039 Å, α = ß = 90 °, γ = 120 ° the grain size ranges from ~ 3-5 microns. Grains are recrystallized grains with boundaries straight or slightly curved, and often there are sub-grains and twinning. Numerous dislocations and precipitates of secondary phases in α - phase were observed.

Materials

602

Frequency dependencies of electrical conductivity of silicon nanoparticles exposed to neutron flux

Elchin Huseynov
National Nuclear Research Center, Inshaatchilar pr. 4, AZ 1073, Azerbaijan
elchin.huse@yahoo.com

It has been reviewed the frequency dependencies of electrical conductivity of nanoparticles affected by neutron flux at different times and initial state, at various constant temperatures such as 100K, 200K, 300K and 400K. Measurements have been carried out at each temperature at the different 97 values of frequency in the 1 Hz – 1MHz range. From interdependence between real and imaginary parts of electrical conductivity it has been determined the type of conductivity. Moreover, in the work it is given the mechanism of electrical conductivity according to the obtained results.
The nanomaterial used in the experiment consists of cubic modification silicon nanoparticles which is has, 80m2/g specific surface area (SSA), 100nm particles' size and 0.08g/cm3 density. Some parameters of oxide form of used samples have been studied analogically [1-5]. It has been reviewed the frequency dependencies of electrical conductivity within initial state and after neutron irradiation. The samples have been irradiated at full power mode (250 kW) by neutron flux (2x1013 n•cm-2s-1) in central channel (channel A1) of TRIGA Mark II light water pool type research reactor in "Reactor Centre" of Jozef Stefan Institute in the city of Ljubljana, Slovenia. The various parameters of neutron flux have been study continuously [6-12]. In the result of comparative analysis of frequency dependences of electrical conductivity of the samples affected by neutron flux with different periods and initial state, it has been established that, clusters are formed inside the sample under neutron flux influence at 100K temperature. As a result of the generated clusters it is observed a sharp chaotization at this temperature. At 200K-300K temperatures the clusters are reduced under the influence of temperature and frequency and fully recombined at 400K. From comparative analysis of frequency dependencies at various temperatures it has been revealed that the value of electrical conductivity increases up to 20 times due to neutron flux influence. From

interdependence of real and imaginary parts of electrical conductivity it has been revealed that under neutron flux influence the real part of electrical conductivity has changed more than the imaginary part. The conductivity to be atomic or dipolar ion type, has been found out from interdependence of real and imaginary parts of electrical conductivity.

References

1. Elchin Huseynov, Adil Garibov, Ravan Mehdiyeva "Temperature and frequency dependence of electric conductivity in nano-grained SiO2 exposed to neutron irradiation" Physica B: Condensed Matter 450, 77 – 83, 2014
2. Elchin Huseynov, Adil Garibov, Ravan Mehdiyeva "Influence of neutron irradiation and temperature on the electric conductivity of SiO2 nanoparticles" Journal of Electrostatics 74, 73–78, 2015
3. Elchin Huseynov, Adil Garibov, Ravan Mehdiyeva, Eršte Andreja, Anar Rustamov "Influence of neutron flux, frequency and temperature to electrical impedance of nano silica particles" AIP Advances 4, 11, 117122 (2014)
4. Elchin Huseynov, Adil Garibov, Ravan Mehdiyeva, Efsane Huseynova "Effects of neutron flux on the nano silica particles: ESR study" Modern Physics Letters B 30, 1650115, 2016
5. Elchin Huseynov, Aydan Garibli "Effect of neutron flux on the frequency dependence of permittivity of nano silicon particles" Silicon, 2015
6. Vladimir Radulović, Žiga Štancar, Luka Snoj, Andrej Trkov, "Validation of absolute axial neutron flux distribution calculations with MCNP with 197Au(n, γ)198Au reaction rate distribution measurements at the JSI TRIGA Mark II reactor", Applied Radiation and Isotopes, Volume 84, 57-65 (2014)
7. Gasper Zerovnik, Manca Podvratnik, Luka Snoj, "On normalization of fluxes and reaction rates in MCNP criticality calculations", Ann. Nucl. Energy 63, 126–128 (2014)

Materials

603

Macroscopic Validation of the Micromechanical Model for Neutron-Irradiated Stainless Steel

Samir El Shawish[1], Leon Cizelj[1], Jeremy Hure[2], Benoit Tanguy[2]

[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia

[2]CEA, Member of SNETP Executive Committee, Gif sur Yvette 91191, France

samir.elshawish@ijs.si

Severe irradiation conditions in nuclear reactors may limit the operational life of internal structural components supporting the reactor core. Neutron irradiation of austenitic stainless steels that constitute internal structures of reactors may results in the deterioration of mechanical and fracture properties. In combination with corrosive environment of the primary water, irradiation induced changes in microstructure and microchemistry may make those steels more sensitive to Irradiation-Assisted Stress Corrosion Cracking (IASCC). IASCC has led to several failures of baffle-to-former bolts in pressurized water reactors, by initiation and propagation of intergranular cracks. The safe operation and reliable structural integrity of power plants require precise prediction of the initiation and propagation of IASCC to establish a strategy for inspection and replacement, especially for long life operation over 60 years. While predictive models of irradiation-induced hardening are available, reliable predictions of IASCC sensitivity are currently unavailable.

Recently, the authors from CEA, France, developed a micromechanical crystal plasticity model to describe a nonlinear mechanical response of austenitic stainless steel subjected to neutron irradiation. The model, based on dislocation dynamics inferred mechanisms and finite strain theory, is able to capture the irradiation-induced hardening followed by softening during plastic deformation. In collaboration between CEA and JSI, the model was used in finite element simulations of realistic stainless steel wire aggregate obtained from X-ray tomography, leading to distributions of stresses at grain boundaries. These local stresses are the driving force of intergranular cracking and need to be accurately determined in order to reliably predict IASCC.

In this study, the existing crystal plasticity model is validated macroscopically through a series of simulation tests of tensile specimens performed on a wide range of deformations and irradiation levels. An emphasis is put on finding and using a converged finite element aggregate model to adjust constitutive law parameters by performing tensile tests up to and beyond necking, where specimen geometry and boundary conditions become increasingly important. In the past, to avoid long simulation times due to complexity of the crystalline law, the initial identification of model parameters was carried out by tensile tests on a simplified, cubic polycrystalline aggregate composed of 343 cubic grains with 1 element per grain. Such an aggregate is too coarse to be a representative volume element of the model. In this work, therefore, an upgraded polycrystalline model is built with real specimen geometry and realistic boundary conditions. To study the effect of grain topology, Voronoi tessellations with different mesh refining are used and compared to simpler cubical grain aggregates. In addition, a novel automatic fitting approach is introduced where model parameters are allowed to take only discrete values in order to speed up the calibration procedure. The adjustment of model parameters is done with respect to measurements on 304L stainless steel.

Materials

604

Drop Test Analysis of Reinforced Concrete Disposal Container

Miha Kramar[1], Franc Sinur[2], Matija Gams[1]
[1]Zavod za gradbeništvo Slovenije, Dimičeva 12, 1000 LJUBLJANA, Slovenia
[2]IBE, d.d., Hajdrihova 4, 1000 LJUBLJANA, Slovenia
miha.kramar@zag.si

The government of Slovenia has decided to build a repository for Low and Intermediate Level Short Lived radioactive waste (LILW) with the capacity of approx. 9400 m3 of LILW. The implementation of the project was assigned to the Agency for Radioactive Waste Management (ARAO). Based on a comparative study of different alternatives, the option with below-ground silos in the vicinity of Krško nuclear plant (NEK) was chosen.

Prior to disposal, LILW will be inserted into concrete disposal containers of type N2b with dimensions of 1,95 x 1,95 x 3,3 m filled with 4 tube-type containers (TCC). The disposal container is qualified as IP-2 package. The transport of containers from NEK to the disposal site will use a public road. Because of the National act on the transport of dangerous goods on public roads, ADR regulations for road transport and manipulation of LILW packages will be followed. ADR requires that containers provide protection against the hazard (radiation) of the material under all conditions of transport, including foreseeable accidents. There should be no more than a 20 % increase in the maximum radiation level at any external surface of the package even in case of the accident. To demonstrate compliance with these requirements a drop test is required: for packages IP-2 with a weight over 15 t a drop test from 0,3 m should be performed.

The design of the container has been carried out by consulting engineering company IBE, d.o.o. which is also responsible for the performing an actual drop test. Different design concepts were considered and their performance to drop tests simulated using numerical simulations. Drop test simulations have been performed at ZAG (Slovenian National Building and Civil Engineering Institute) with a general purpose finite element program Abaqus using explicit dynamics procedure (Abaqus/Explicit). The container has been modelled in detail with 3D solid elements. Nonlinear material properties were considered while multiple contact surfaces were assumed between different parts of the container (e.g. between the lid and the container, etc.). The reinforcement was included in the model (in the form of embedded nonlinear beams). Different drop scenarios were investigated exceeding the requirements of ADR: a drop from 0,3 m onto the most vulnerable corner was simulated as well as the overturning of a container which might follow the initial collision. The accuracy of the numerical results was checked by controlling the impact energy flow and performing sensitivity analyses to different parameters (contact properties, type of elements, stiffness of the base, etc). Special attention was devoted to establishing quantifiable acceptability criterion based on different output variables such as crack widths, deformations and cumulative damage.

The analyses have shown that overturning of container might be more critical than the drop on the corner which causes mostly local damage in concrete. In some cases the damage predicted in the numerical analysis was substantial indicating that the container might not fulfil the ADR requirement. Therefore, it was necessary to change the design of the container. The design of the container is still ongoing and other alternatives will be tested. Finally, an actual drop test will be performed for the validation of the numerical model and implementation of even more robust analyses.

701

Shutdown Probabilistic Safety Assessment – A Case Study for the Pressurized Water Reactor

Marko Čepin[1], Rudolf Prosen[2]
[1]Fakulteta za elektrotehniko, Tržaška cesta 25, 1000 Ljubljana, Slovenia
[2]Nuklearna elektrarna Krško, Vrbina 12, 8270 Krško, Slovenia
marko.cepin@fe.uni-lj.si

Shutdown probabilistic safety assessment may be considered as an extension of power probabilistic safety assessment applied to the shutdown conditions of the nuclear power plant. It is more complex in sense of variety of models, because the plant shutdown is a sequence of plant operating states, which differ among each other so much that the probabilistic safety assessment models, which represent them, need to be varied in order to represent them realistically. The method is developed, which follows mostly the steps known in power probabilistic safety assessment with an exception of definition of plant operating states, which are defined in a way that one plant operating state suits its respective specific probabilistic safety assessment model. The number of plant operating states has to be large enough to consider the differences between them in sense of the related plant parameters and configurations of the plant equipment. At the same time, the number of plant operating states has to be small enough, in order that the number of the probabilistic safety assessment models representing them is small enough for the performance and cost effectiveness of the related work. The results of a case study based on a two loop nuclear power plant with the pressurized water reactor are presented and discussed. The results show that the risk of different plant operating states varies significantly and is in average lower than the risk of the plant in full power operation.

Probabilistic Safety Assessment

702

Challenges of external hazards assessment. ASAMPSA_E project achievements.

Mirela Nitoi[1], Emmanuel Raimond[2], Yves Guigueno[2]
[1]RATEN ICN Institutul de Cercetari Nucleare Pitesti, Str. Campului nr.1, 115400 Mioveni, Romania
[2]IRSN - Institut de radioprotection et de sureté nucléaire, Nuclear Safety Division, BP17, 92262 Fontenay-aux-Roses Cedex, FRANCE, France
mirela.nitoi@nuclear.ro

One of the main challenges for the nuclear domain is represented by an efficient management of the critical events, events that have an impact on the safety operation of nuclear power plants (NPP).
The external hazards constitute a significant source of perturbations in the safe operation of a NPP, and for this reason, to properly investigate them and to find ways to obtain a better picture about their consequences, is quite important.
The Advanced Safety Assessment Methodologies: extended PSA (ASAMPSA_E) project aims to examine in detail how efficient is the Probabilistic Safety Assessment (PSA) methodology in identifying any major risk induced by the interaction between a NPP and its environment, and to develop some guidance documents and technical recommendations for PSA developers and users, dedicated to improve the quality of the PSA studies. Launched after the Fukushima accident, the ASAMPSA_E project pays an increased attention to the risks induced by the occurrences of external events and their combinations.
The paper presents the work packages and the organizations involved as partners in the project. The main directions of actions, the identified priorities for the project and the challenges encountered in the attempt to assess the external

hazards are discussed. The results obtained so far and the recommendations developed in frame of the project, regarding the assessment of external hazards, are specified. The connection with the stakeholders and their expectations in terms of guidance for the development and use of extended PSA are highlighted.

Probabilistic Safety Assessment

703

Assessments of EP&R provisions in Europe

Nadja Železnik
Regionalni center za okolje za srednjo in vzhodno Evropo , Slovenska cesta 5, 1000 Ljubljana, Slovenia
nzeleznik@rec.org

The Fukushima accident in March 2011 has intensified European concerns about off site nuclear emergency preparedness and response. As this important aspect of defence in depth was not included in the EC/ENSREG process of stress tests, several initiatives took place afterwards. The HERCA association formed a working group on “Emergencies” and started to work on the proposition leading to a uniform way of dealing with any serious radiological emergency situation, regardless of national border lines. In 2013 DG ENER commissioned a “Review of current off-site nuclear emergency preparedness and response arrangements in EU member States and neighbouring countries” which provided the evaluation of the EU EP&R provisions. In parallel a civil society association Nuclear Transparency Watch has organised an assessment of EP&R provisions across the Europe from civil society point of view and reported findings.
The findings of all investigations show that current arrangements and capabilities for off- site nuclear EP&R appear, on paper, to be broadly compliant with current EU legislative requirements and international guidance. However, more deep examinations of arrangements in practice identified a number of gaps and inconsistencies that need to be addressed, like not harmonised criteria and cross -border arrangements, mainstreaming of nuclear emergency preparedness into civil protection mechanisms, long term protective measures and strategies, involvement of local population and communication, inclusion of societal development (new social media, new spatial and demographic development,…). New Basic Safety Standard directive, adopted in 3013, and addressing also EP&R requirement could be a good opportunity to improve the EP&R arrangements if not taken only formally. The paper will present the findings of different investigations and recommendation for improvement of EP&R.

801

CoreSOAR Update of the Core Degradation State-of-the-Art Report: Status September 2016

Tim Haste[1], Marc Barrachin[1], Georges Repetto[2], Martin Steinbrück[3], Paul David William Bottomley[4]

[1]Institut de Radioprotection et de Sureté Nucléaire, Bât. 702 Centre de Cadarache, BP 3-13115 Saint Paul lez Durance, France

[2]Institut de Radioprotection et de Sureté Nucléaire (IRSN) Centre d' Etrudes de Cadarache, Cadarache B.P 3, Batiment 702, F-13115 Saint Paul-les-Durance CEDEX, France

[3]Karlsruhe Institute of Technology, P.O. Box 3640, 76021 Karlsruhe, Germany

[4]JRC-Directorate for Nuclear Safety and Security , Hermann-von-Helmholtz-Platz 1, P.O. Box 2340,, 76125 Karlsruhe, Germany

tim.haste@irsn.fr

In 1991 the Commitee on the Safety of Nuclear Installations (CSNI) published the first State-of-the-Art Report on In-Vessel Core Degradation, which was updated to 1995 under the European Union (EU) 3rd Framework Programme. These covered phenomena, experimental programmes, material data, main modelling codes, code assessments, identification of modelling needs, and conclusions including the needs for further research. This knowledge is relevant to such safety issues as in-vessel melt retention of the core (IVMR), recovery of the core by water reflood, hydrogen generation and fission product release.

In the following 20 years, there has been substantial progress in understanding, with major experimental programmes finished, such as the integral in-reactor Phébus FP tests, and others with many tests completed, e.g. in the integral ex-reactor QUENCH series on reflooding degraded rod bundles, and LIVE, on melt pool behaviour in the lower head. These are accompanied by separate-effects tests to study individual phenomena in more detail. A similar situation obtains regarding integral modelling codes such as MELCOR (USA) and ASTEC (Europe) that encapsulate current knowledge in a quantitative way. After the two EU-funded projects on the SARNET network of excellence, now continuing in the NUGENIA association, it is timely to take stock of the knowledge gained.

The CoreSOAR project, in the NUGENIA/SARNET framework, draws together the experience of 11 European partners with the aim of comprehensively updating the state of the art in core degradation, over the next two years. The review of available data has begun, and this paper as an example indicates recent progress in small-scale tests involving material interactions that provide data to cover the gaps in knowledge identified in the integral tests such as Phébus FP. This is complemented by detailed examination of post-test samples to elucidate the mechanisms involved. It also considers advances in thermodynamic databases necessary to model the formation and relocation to the lower head of the complex corium compositions involved, as well as phenomena related to in-vessel retention. The report will serve as a reference point for ongoing research programmes in NUGENIA, in other EU research projects such as in Horizon2020, and in CSNI, e.g. the Fukushima benchmark BSAF.

802

Nordic collaboration: Impact of Ag and NOx compounds on the transport of ruthenium in the primary circuit of NPP in a severe accident

Teemu Kärkelä[1], Ivan Kajan[2], Unto Tapper[1], Leena-Sisko Johansson[3], Melany Gouello[4]
[1]VTT Technical Research Centre of Finland, Tietotie 3, Espoo, 02044 VTT, Finland
[2]Chalmers University of Technology, Kemirägen 4, SE-41296 Goeteborg, Sweden
[3]Aalto University, School of Science, P.O. Box 11000, 00076 Aalto, Finland
[4]VTT, Tietotie 3, FI-02150 Espoo, Finland
teemu.karkela@vtt.fi

During the operation of a nuclear power plant (NPP), a significant amount of ruthenium is built up in the fuel as a product of the nuclear fission. The importance of ruthenium from the radiological point of view is mainly due to the isotopes 103Ru and 106Ru with half-lives of 39.35 days and 373.5 days, respectively. When ruthenium is released from the fuel to the environment in a severe NPP accident, these ruthenium isotopes cause a radiotoxic risk to the population both in a short and long term by building-up to the human body and external exposure to the radiation, thus possibly leading to a development of cancer.

The transport of ruthenium through a reactor coolant system (RCS), after being released from the fuel, has been investigated in several experimental programmes recently. The VTT Ru transport programme has shown that the release of Ru from RuO2 powder was dependent on the oxygen partial pressure in air-steam atmospheres at 827, 1027, 1227 and 1427 °C. The highest fraction of gaseous RuO4 at the outlet of the model primary circuit was observed at 1027 °C oxidation temperature. At higher temperatures, ruthenium transported mainly as RuO2 aerosol. In the experiments of RUSET programme it was observed that the presence of other FPs, e.g. BaO and CeO2, as mixed with the metallic Ru precursor when the sample was oxidized at 1100 °C, decreased the fraction of gaseous RuO4 in the outlet air over the stainless steel surface compared to the pure Ru oxidation. It was also shown that the transport of RuO4 was dependent on the surface material in the coolant circuit. In both VTT and RUSET programmes it was noticed, that the partial pressure of RuO4 reaching the outlet of model primary circuit was in the range of 10-7 to 10-6 bar, which is significantly higher than what is expected based on thermodynamic equilibrium calculations.

As the previous studies have mainly been conducted in pure air-steam atmospheres, the current study was dedicated to air ingress conditions with representative airborne fission product/control rod (Ag) and air radiolysis (NOx) species which were mixed with vaporized Ru oxides. The aim was to study the impact of these additives on the transport of ruthenium as gas and particles through the primary circuit of nuclear power plant in a severe accident. As a main outcome, the transport of gaseous ruthenium through the facility increased significantly when the oxidizing NO2 gas was fed into the atmosphere. The feed of pure silver particles into the gas flow showed a significant decrease in gaseous RuO4 reaching the outlet of the facility. Simultaneously, a noticeable increase of ruthenium in form of RuO2 trapped on the filter was observed. When both silver aerosol and NO2 in form of AgNO3 compound were fed into the atmosphere, the transport of ruthenium in gaseous and aerosol forms was promoted. Based on experiments it was concluded that the composition of atmosphere in the primary circuit will have a notable effect on the speciation of ruthenium transported into the containment building during a severe accident.

803

Investigation of external reactor pressure vessel cooling with ATHLET-CD

Peter Pandazis[1], Sebastian Weber[2]
[1]Gesellschaft für Anlagen- und Reaktorsicherheit Forschungsgelände, Postfach 12-28, 85748 Garching b. München, Germany
[2]Gesellschaft für Reaktorsicherheit (GRS), Schwertnergasse 1, 50667 Köln, Germany
peter.pandazis@grs.de

Severe accident management (SAM) strategies are coming into the focus of nuclear safety investigations after the Fukushima accident. The main goal of these strategies is to prevent the release of radioactive fission products in the environment. During a severe accident in light water reactors after SCRAM the fuel elements and reactor internals may start to melting due to the decay heat if the cooling systems fail. The collected molten fuel and core internals (corium) relocate into the lower plenum after the collapse of the grid plate (Pressurized Water Reactor, PWR) or the control rod guide tubes (Boiled Water Reactor, BWR) because of the mass and high temperature of the corium. Without any further counter-measure the thermo-chemical attack of the corium leads to a melt through of the RPV wall and radioactive materials release into the environment.
The SAM strategy of in-vessel retention by ex-vessel cooling has been developed to minimize the risk of fission product releases into the containment. According to this strategy the corium will be stabilized within the lower plenum by transferring the decay heat through the wall into the containment via external cooling. The external cooling of the RPV is realized via flooding of the reactor cavity. Comprehensive works showed that the success of this SAM concept depends mainly on the inner thermal load (mass, composition and decay heat of corium) and on the coolability of the wall (cooling channel shape, massflow, roughness, etc.). In this work a method has been developed and adopted, using the thermal-hydraulic system code ATHLET-CD (Analysis of Thermal-hydraulics of Leaks and Transients with Core Degradation) developed by GRS, to investigate the SAM concept of in-vessel melt retention (IVMR) by ex-vessel cooling.
The investigations have been performed using a German generic BWR design and include the simulations of the transient corium behavior, the structural response of the wall as well as the external cooling process. In the simulations a Station Black Out accident scenario has been assumed in addition to the failure of the core cooling systems. The key-points of successful cooling have been determined with the developed model considering the actual geometry and accident scenario. Furthermore, the calculations have been demonstrated the applicability of ATHLET-CD to perform complex and effective analyses to evaluate the SAM strategy IVMR.

804

Comparison and analysis of corium pool behavior in lower head modeled by MAAP (EDF version) and PROCOR (CEA) codes

Sophie Bajard[1], Nikolai Bakouta[2], Benoit Habert[1], Romain Le Tellier[1], Laurent Saas[1]
[1]CEA, Member of SNETP Executive Committee, Gif sur Yvette 91191, France
[2]Electricite de France, 140 Avenue VITON, 13009 Marseille Cedex 20, France
sophie.bajard@cea.fr

Corium pool formation in the lower head is likely to occur during a Severe Accident (SA) in Light Water Reactors (LWR). In this situation, the thermal load on the vessel is largely dependent on the molten pool stratification in terms of the immiscible oxide and metal phases.
According to the results given by some experimental programs (RASPLAV, MASCA, CORDEB), the number and the positions of the layers in the corium pool can evolve within a transient process involving chemical elements migration. In order to describe this phenomenology, in-vessel corium models have been adapted in PROCOR (CEA) and modified from the original MAAP models, in MAAP4* and MAAP5* (EDF proprietary version). So, these codes share important common features:
- 0D mass and energy conservation equations for the corium pool layers;
- a kinetic inter-layer mass transfer model based on a simplified representation of the miscibility gap in oxide-metal corium systems and controlled by the Uranium diffusion in the oxide phase;
- chemically reacting metal and oxide layers surrounded by an oxide crust;
- a steel layer above the oxide crust.

With the models complexification, the need for code verification and validation increases. In this framework, detailed code cross-comparisons of the most relevant models with increasing complexity provide a helpful approach. Accordingly, several numerical benchmarks have been constructed by EDF and CEA regarding in-vessel corium transient modelling

A first set of three calculations have been performed with MAAP4-EDF and PROCOR, starting from the simplest (thermal exchanges of a homogeneous corium pool) to the most complex (thermochemical and thermal models with a three layer pool). For each benchmark, the differences have been understood and some specific adjustments (user parameters for example) were introduced in the codes, in order to lead to the best estimated results.
A second set of two calculations have then been performed with MAAP4*, but also MAAP5* and PROCOR, in order to cover all the possible corium pool stratifications, with finally a transient one which can be representative of a reactor case.

In this paper, first we briefly summarize the results given by the first set of calculations, which have already been presented in ERMSAR 2015 (in Marseille, France). The causes of the differences were mainly due to the equation of state associated to the corium (linking its enthalpy H to its temperature T) and to the form of the energy conservation equation, written in enthalpy in MAAP whereas it is written in temperature in PROCOR. Then, the heat transfer correlations and the physical properties data also showed some discrepancies. On this basis, we detail two additional benchmarks that have been completed recently. In particular, in the last benchmark, a more complex transient of the corium pool in the vessel lower head has been investigated. This case is more representative of a reactor case and involves:
- a transient configuration with a three layer corium pool simulating the stratification inversion process;
- a transient migration of the upper steel layer towards the melt pool, through the upper surrounding crust.

These benchmark activities have helped both CEA and EDF to improve their numerical tool by a better understanding of the models features and of their coupling impact on some typical scenarios. It has also helped to define the most important R&D tasks to be carried out for their enrichment or improvement.

MAAP4* / MAAP5*: MAAP4 or MAAP5 version, modified by EDF for R&D needs

805

Detailed Thermal-Mechanical Modelling of Cylindrical Core Support Plate During Severe Accident in PWR

Maciej Skrzypek, Eleonora Klara Skrzypek
National Centre for Nuclear Research, ul. Andrzeja Sołtana 7, Otwock-Świerk, Poland
maciej.skrzypek@ncbj.gov.pl

In the field of severe accident analysis of nuclear reactors many calculations focus on corium propagation. Molten material can drain laterally or axially to the lower plenum of Reactor Pressure Vessel (RPV). In case of axial draining corium can be kept on the core support plate or drain through, depending on the thermal and mechanical loads (strains). Detailed calculations in this area are necessary to precisely predict possible rupture of the vessel and time, because any experimental data exist. Those parameters can be calculated using Finite Element Method which can help to obtain a database of certain analysis.
Implementation of the plate model to fast running parameter code allows to better understand phenomena under severe accidents and determine rupture of the vessel, taking into consideration also mechanical analysis, which is very often omitted by code developer.

Severe Accidents

806

Influence of Melt Pouring on Stratified Steam Explosion

Vasilij Centrih, Matjaž Leskovar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
vasilij.centrih@gmail.com

A steam explosion may occur during a severe reactor accident when the molten core comes into contact with the coolant water. An important condition for the occurrence of a steam explosion is the initial coarse premixing of the melt and the water. In nuclear reactor safety analyses steam explosions are primarily considered in the melt jet-water pool configuration, where due to the melt jet fragmentation the required premixture is efficiently produced. In stratified melt-water configuration, i.e. molten corium layer below water layer, it was previously assumed that there is no premixing of the melt and the water and that a strong explosion thus cannot develop. The recently performed experiments in the PULiMS and SES (KTH, Sweden) facilities with corium simulant materials however revealed that strong steam explosions may develop spontaneously also in stratified melt-water configuration. The development of a considerable melt-water premixed layer above the spread melt was clearly visible in the tests where an explosion occurred. In these experiments the melt was poured into a shallow pool of water. Despite the shallow water, the melt jet fragmentation during the short penetration through the water may possibly importantly influence the formation of the premixture also in such a stratified configuration. To address this issue an underwater melt release stratified steam explosion experiment is planned to be performed in the SES facility in the frame of the EC SAFEST project.

In the paper the potentially important influence of the melt pouring on the stratified steam explosion will be studied by computer simulations with the MC3D code in the SES geometry. Since no validated models have been developed yet which would adequately describe the observed underwater melt spreading and the formation of the observed premixed layer in stratified configuration, an innovative approach for the coupling of the premixing phase calculation, the premixture layer characteristics and the explosion phase calculation with available MC3D code procedures was established. The performed

comparative analysis for two scenarios, first one with underwater melt release and the second one with the melt pouring above the water, will be presented and discussed. Also suggestions for further experimental and analytical work will be given.

Severe Accidents

807

Simulation of natural circulation experiment in MISTRA experimental containment facility with OpenFOAM CFD code

Boštjan Zajec, Ivo Kljenak
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
ivo.kljenak@ijs.si

Various experiments on the behaviour of non-homogeneous atmosphere are being performed in containment experimental facilities, both to understand the phenomena and to obtain results, adequate for validating Computational Fluid Dynamics (CFD) codes for the purpose of using them to simulate phenomena in actual plants. Within the OECD project SETH-2, which lasted from 2007 to 2010, experiments have been performed in the MISTRA (Commissariat a l'Energie Atomique et aux Energies Alternatives, Saclay, France) and PANDA (Paul Scherrer Institute, Villigen, Switzerland) containment experimental facilities.

One of the experiments in the MISTRA facility (which is a cylindrical vessel with a volume of 98 m2, with some internal subdivisions), called NATHCO, consisted in gradually heating condensers, installed near the vessel wall, so as to heat the nearby gas and induce buoyant flow in the stagnant atmosphere. The influence of this natural circulation on a previously established horizontal layer of helium in the upper part of the vessel was observed.

The experiment NATHCO was simulated with the open-source CFD code OpenFOAM. A two-dimensional axisymmetric model of the MISTRA facility was developed. Simulation results (time-dependent local helium concentrations and temperatures) are compared to experimental data and the discrepancies are analysed.

Severe Accidents

808

Modelling of debris bed coolability in bottom reflooding conditions with MC3D code

Janez Kokalj, Mitja Uršič, Matjaž Leskovar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
kokalj.janez@gmail.com

A hypothetical severe accident in a nuclear power plant has the potential for causing severe core damage, including meltdown. To prevent or in the case of already formed debris bed to limit the in-vessel core degradation the basic severe accident management strategies consider the in-vessel reflooding to ensure the coolability. Due to the debris bed porosity, which allows easier coolant intrusion, the debris bed provides greater chances for cooling than a pool of molten corium. When the cooling is not sufficient, with the continuation of the scenario the degraded reactor core is melted and relocated to the lower reactor vessel plenum. To prevent the ex-vessel melt release the in-vessel melt retention strategy could be applied.

The coolability of the debris bed was recognized as an important nuclear safety issue in the frame of the EU SARNET-2 (Severe Accident Research NETwork of Excellence) programme. In the SARNET-2 programme the ex-vessel debris bed formation due to the fuel-coolant interaction and the coolability of the formed debris bed were analysed. Currently the international research on the debris bed coolability is under investigation in the frame of the Horizon 2020 IVMR (In-Vessel Melt Retention Severe Accident Management Strategy for Existing and Future NPPs) project.

The purpose of our research is to understand the key processes related to the in-vessel debris coolability in the bottom reflooding conditions. First, the recently performed tests in the PEARL facility (IRSN, France) will be presented. The PEARL experimental program was launched to provide experimental data to validate 2-D and 3-D models for the debris bed bottom reflooding. Next, the modelling and analysis of the PEARL experiments using the MC3D code (IRSN, France) will be described. The aim of the performed work was to analyse the uncertainties in the initial experimental conditions and to assess the heat transfer modelling approach in the MC3D code.

Severe Accidents

809

Comparison of CFD and LP Codes for the Simulation of Hydrogen Combustion Experiments

Tadej Holler[1], Ed Komen[2], Ivo Kljenak[1]

[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia

[2]NRG-Nuclear Research and Consultancy Group Dept. Fuels, Actinides and Isotopes, P.O.Box 25, 1755 ZG Petten, Netherlands

tadej.holler@ijs.si

The production and release of hydrogen into the containment during a severe accident is an important safety issue for Light Water Reactors (LWRs). Combustion of hydrogen may cause structural damage to the containment and may compromise its function as final barrier for release of radioactive fission products to the environment. To reduce the hydrogen risk as far as possible, hydrogen mitigation systems such as Passive Auto-catalytic Recombiners (PARs) and igniters can be installed. The risk of hydrogen deflagration has received increased attention after the Three Mile Island accident in the USA back in 1979, and also most recently following the Fukushima accident in Japan in 2011, where hydrogen's destructive power was displayed. Computational modeling is required to demonstrate the adequacy of the Nuclear Power Plant's (NPP's) hydrogen risk management systems as well as for their optimal design and the assessment of the accompanied residual risks of the presence of hydrogen.

Historically, so-called system codes or lumped parameter codes were used for the assessment of hydrogen deflagration risk in NPPs. With the advancement of computers and thus increase of computational power in recent time, complementary to the use of lumped parameter codes, Computational Fluid Dynamics (CFD) modeling can be used for more detailed assessment of the hydrogen risks in determining the possibility of a breach of the NPP's containment integrity.

This paper presents comparison of simulation results obtained using the Ansys Fluent CFD code and the lumped parameter code ASTEC (Accident Source Term Evaluation Code). It offers brief theoretical backgrounds of both modelling approaches and also focuses on advantages and drawbacks of both applied methods for the use in hydrogen combustion risk assessment.

Three experiments performed with hydrogen-air mixtures and different initial hydrogen concentrations in the medium-scale THAI experimental facility were used for the presented comparative analysis. A detailed analysis and comparison also with the experimental results of maximum pressure are presented along with a discussion about the future of computational analyses in the field of hydrogen combustion safety in NPPs' containments.

810

Improvement of the melt relocation modelling in ATHLET-CD

Liviusz Lovasz[1], Sebastian Weber[2]
[1]Gesellschaft für Reaktorsicherheit mbH, Forschungsgelände, 85748 GARCHING, Germany
[2]Gesellschaft für Reaktorsicherheit (GRS), Schwertnergasse 1, 50667 Köln, Germany
liviusz.lovasz@grs.de

The accident in Fukushima pointed out the importance of severe accidents simulations. The severe accidental phenomena are not fully understood due to the lack of experiments, therefore the continuous development of the tools capable of simulating severe accidents is important.
The system code ATHLET-CD (Analysis of THermal-hydraulics of LEaks and Transients with Core Degradation) is designed to describe the reactor coolant system thermal-hydraulic response during severe accidents, including core damage progression as well as fission product and aerosol behaviour, to calculate the source term for containment analyses, and to evaluate accident management measures. The ATHLET-CD structure is highly modular in order to include a manifold spectrum of models and to offer an optimum basis for further development.
In the module ECORE, which simulates the degradation phenomena, the core is divided in concentric rings, the fuel and absorber rods in every ring are modelled by a representative rod. The melt relocation is simulated by rivulets with constant velocity and cross section (candling model), starting from the node of rod liquefaction. The movement of these rivulets is simulated only in axial direction in every ring in the last released version of ATHLET-CD.
Developments were performed to achieve radial relocation of the melt in case of blockage formation. Due to these developments the code is capable of comparing the heights of melt rivulets in neighbouring rings above a blockage and of calculating the radial movement of the molten fuel, based on the height differences of the melt rivulets. The new model takes also the BWR specific elements into account.
To demonstrate the effects of the development, a simulation with the improved and with the original version of ATHLET-CD is presented on an example of a hypothetical severe accident in a generic BWR reactor, with an initial event of a Station Blackout.

Severe Accidents

811

Analysis of X-Ray Images in SERENA KROTOS Experiments with Premixing Simulations

Vasilij Centrih, Matjaž Leskovar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
vasilij.centrih@gmail.com

A steam explosion is an energetic fuel-coolant interaction process, which may occur during a severe reactor accident when the molten core comes into contact with the coolant water. An important condition for the occurrence of a steam explosion is the initial coarse premixing of the melt and the water. The most distinctive process is the melt jet fragmentation and the corresponding premixture formation. To resolve the open issues in steam explosion understanding in the field of nuclear safety a number of activities were carried out within the recent OECD SERENA project. Lately, the comprehensive summary of the post processed x-ray radioscopy data analysis (CEA, France) was completed for the SERENA KROTOS experiments. The analysis of the x-ray images provides an additional insight into the complex premixing processes.
In the paper the new information about the premixture formation provided by the innovative x-ray radioscopy system will

be studied by computer premixing simulations with the MC3D code. Because the local data of the premixing region may be obtained from the x-ray images, the study is focused mainly on the lateral distribution of the premixture along the test section. The analysis is based on the most informative x-ray data from the KS-1 and KS-4 experiments. A parametric analysis was performed, varying the experimental conditions, material properties and some model specific parameters such as the lateral velocity of the produced droplets from jet the fragmentation. Especially the influence of the subcooling and the corium material on the premixture formation will be analysed and discussed. Also suggestions for further experimental and analytical work will be given.

Severe Accidents

813

Material Influence on Ex-vessel Steam Explosion

Tomaž Skobe, Matjaž Leskovar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
tomaz.skobe@ijs.si

A steam explosion may occur, when during a severe reactor accident the corium melt comes into contact with the coolant water. Steam explosions are an important nuclear safety issue because they can potentially jeopardize the primary system and the containment integrity of the nuclear power plant.
In the paper the material influence of the oxide and metal corium on the ex-vessel steam explosion will be presented. A PWR ex-vessel steam explosion study with oxide and metal corium was carried out with the MC3D code. For each type of the corium a premixing simulation and an explosion simulation was performed, triggering the explosion at the time of melt bottom contact. The premixing simulations were performed with the global jet breakup model. Comparing calculations with oxide and metal corium were performed without oxidation in the premixing and the explosion phase. With the comparison of the oxide and metal corium simulation results the influence of the material properties of the melt on the strength of the steam explosion was analysed.

Severe Accidents

814

Thermal-Hydraulic Analysis of PHWR Containment using MELCOR Code in severe accident

Sungchu Song, Seon Oh Yu
Korea Institute of Nuclear Safety, 34 Gwahak-ro, Yuseong-gu , Daejeon 305-338, South Korea
scsong@kins.re.kr

As a major safety function, a containment of Nuclear Power Plant (NPP) provides a reliable mean to confine possible fission products release during severe accident. However, the severe accident occurred at Fukushima NPPs raised the concern on the integrity of containment building, which was failured accompanied by the severe accident. Thus it is of essence to guarantee the integrity of containment building for any type of commercial nuclear systems of Boiling Water Reactors (BWRs), Pressurized Water Reactors (PWRs), and Pressurized Heavy Water Reactors (PHWR). Variuos systems codes of ASTEC, MAAP, and MELCOR have been developed for the purpose and among them the MELCOR code developed for severe accident analysis of PWR and BWR shows the capability of detailed thermal-hydraulic behavior inside the PHWR containment building. In the present work, the PHWR containment with volume of 48,000 m3 was modelled with 52 control volumes of containment components, environment with 1 control volume and 118 flow paths. To quantify the overall risk

on the containment integrity, the H2 concentration, which is the dominant combustible gas, was evaluated. In addition, the effectiveness of various Engineering Safety Features (ESFs) such as PAR, spray system, igniter, and local air cooler has been investigated given the postulated severe accident scenario using the MELCOR code. The time of containment failure caused by increased concentration of hydrogen and the overpressurization has been also evaluated quantitatively to provide important scoping time for the accident management.

901

SOLPS-ITER Dashboard

Leon Kos[1], Ivan Lupell[2], Xavier Bonnin[3]
[1]University of Ljubljana, Faculty of Mechanical Engineering, LECAD Laboratory, Aškerčeva cesta. 6, 1000 Ljubljana, Slovenia
[2]EUROfusion Consortium, JET, Culham Science Centre, OX14 3DB, Abingdon, United Kingdom
[3]ITER Organization, Cadarache Centre, Building 519, 13108 St. Paul lez Durance, France
leon.kos@lecad.fs.uni-lj.si

The design of the ITER divertor and estimates of the required fuelling throughput have relied for many years on simulations performed by use of the SOLPS plasma edge modelling tool. The newly developed SOLPS GUI is a framework tailored specifically for the SOLPS-ITER code suite in a sense that code specifics are built into the interface. Its design allows users to extend functionality by coupling custom widgets prepared for the SOLPS GUI. These custom widgets are in similar environments called actors as they do act on some data, depending on input received and then they pass results further in a scientific workflow. Custom widgets for SOLPS are operating in a similar fashion in a way that they receive and send the signals to other widgets for further operation. In principle, no programming is needed by users to create their own “Dashboard” for analysing and controlling the SOLPS simulations.
In contrast to scientific workflow engines such as Kepler here we are more oriented to look-and-feel experience than to create a general purpose workflow engine. That’s why the widgets in the SOLPS GUI are designed to have “nice” input and output presentation while we don’t care how “nicely” wires are placed. “Wiring” is usually taking significant space in other workflow engines where actors are “small” or have a unified size with separated or neglected display output. The SOLPS GUI uses the reverse approach with widgets filling up the available Dashboard window completely. There can be many widgets that trigger part of the workflow, whereas there are just play/pause/stop buttons used in
Kepler. The SOLPS GUI signal/slot philosophy provided by Qt framework is similar to input/output ports in Kepler, while the triggering is more explicit than implicit. Users are therefore encouraged to design their own Dashboard
by redesigning it to suit their needs. As the dashboard is intended to be configured with Qt designer this means that all actions needs to be provided within the widgets and connected by signals. “Wiring” can be graphical too. The GUI is then saved in XML files and compiled on-the-fly at the GUI startup. Even when providing a limited set of “custom” widgets, there can exist many different dashboards for running SOLPS simulations. They may differ on the analysis, user’s preferences and may be exchanged for reuse by others. Presented graphical programming is not just adding functionality easily but one can simply/remove the unwanted custom widgets and further extend it to other simulation code suite.

902

Calculations to support JET neutron yield calibration: Effects of the neutron source anisotropy

Aljaž Čufar[1], Paola Batistoni[2], Igor Lengar[1], Sergey Popovichev[3], Luka Snoj[1], JET Contributors[4]

[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]ENEA Fusion Division, Via E. Fermi 45, 1-00044 Frascati, Rome, Italy
[3]Culham Centre for Fusion Energy, Abingdon, Oxon, OX14 3DB, United Kingdom
[4]EUROfusion Consortium, JET, Culham Science Centre, OX14 3DB, Abingdon, United Kingdom
aljaz.cufar@ijs.si

The calibration of JET's main neutron detectors, fission chambers and activation system, is based on the measurements of the detector response to a calibration neutron source placed in multiple positions inside the vacuum vessel. The main requirement for the calibration neutron source are well known source characteristics such as the source intensity and energy spectrum as well as its anisotropy. The information about these parameters is crucial if the JET's neutron detectors are to be calibrated with the target accuracy of 10 %. These neutron source characteristics are obtained through a combination of Monte Carlo simulations and characterisation measurements performed at a neutronics laboratory.

When an anisotropic neutron source is put into a complex geometry, such as the tokamak's vacuum vessel, various complications can arise. Small changes in the position or orientation of the source can significantly affect the detector response, depending on the anisotropy profile and detector's response function. Monte Carlo simulations are an important tool in the preparatory phase to the experimental in-situ calibration as it is important to understand the difficulties and possible sources of errors before the calibration experiment is performed.

For the calibration of JET's neutron detectors to neutrons with energy of 14 MeV a compact accelerator based DT neutron generator will be used as a calibration source. Such a generator is inherently anisotropic while additional anisotropy is introduced as a result of the materials surrounding the area where neutrons are produced. Small changes in the position or orientation of the source, when positioned inside the tokamak, can lead to significant change of the detector responses. This influences the accuracy of the calibration process. The investigation of the effects of uncertainties in the generator's position and orientation will be presented along with their effects on the accuracy of the calibration.

903

Fast online MPC for ITER plasma current and shape control

Samo Gerkšič

Institut Jožef Stefan, Jamova cesta 39, 1000 Ljubljana, Slovenia
samo.gerksic@ijs.si

In a tokamak reactor, the Plasma Current and Shape Controller (PCSC) is the component of Plasma Magnetic Control (PMC) that commands the voltages applied to the poloidal field coils, to control the coil currents and the plasma parameters, such as the plasma shape, current, and position. The PCSC acts on the system pre-stabilised by the Vertical Stabilisation controller. The task of PMC is to maintain the prescribed plasma shape and distance from the plasma facing components, in presence of disturbances, e.g. H-L transitions or ELMs, subject to changes of local dynamics in different operating points.

Model Predictive Control (MPC) is considered as the most important technique in advanced process control technique in the process industry. It has gained wide industrial acceptance by facilitating a systematic approach to control of large-scale multivariable systems, with efficient handling of constraints on process variables and by enabling plant optimisation. These advantages are considered beneficial for PCSC, and potentially also for other control systems of a tokamak. The main obstacle to using MPC for control of such processes is the restriction of the most relevant MPC methods to processes with relatively slow dynamics due to the relatively long achievable sampling times, because time-consuming on-line optimization problems are being repeatedly solved at each sample time of the CSC control loop for determining control actions.

In this work we explore the practical feasibility of using MPC for PCSC in the ITER tokamak, employing recently developed fast on-line quadratic programming (QP) optimization methods with complexity reduction techniques. A survey of the available QP methods suitable for the on-line solution of MPC optimization problems is given, with emphasis on first-order methods, which have been recently considered as the most promising candidates for fast online MPC control. The prototype MPC controller [1] is based on the control scheme of [2]. Using a modification of the QP solver QPgen [3], a five-fold speed-up compared to the state-of-the-art commercial solver CPLEX was achieved, with peak computation times less than 10 ms on a computer with a four-core Intel processor running real-time Linux. This is already considered sufficiently fast for the 100 ms sample time estimated to be suitable for the ITER CSC control loop.

[1] S. Gerkšič, G. De Tommasi, "Model predictive control of plasma current and shape for ITER", 28th Symposium on Fusion Technology (SOFT 2014), San Sebastián, Spain
[2] G. Ambrosino et al., IEEE Trans. Plasma Science, 37(7), 2009, 1324-1331
[3] Giselsson P., Improving Fast Dual Ascent for MPC - Part II: The Embedded Case, arXiv (2014)

Nuclear Fusion and Plasma Technology

904

CAD data storage and access in IDAM

Marijo Telenta[1], Leon Kos[1], Robert Akers[2]
[1]University of Ljubljana, Faculty of Mechanical Engineering, LECAD Laboratory
[2]EUROfusion Consortium, JET, Culham Science Centre

Integrated Data Access Management (IDAM) is data access tool for analysis, visualisation, and modelling. It is developed at CCFE for MAST-U data access. Data access is based on data objects from within the files. IDAM is also put capable. Metadata from raw and analysed files is written to the IDAM database. IDAM server uses a plugin architecture for each data resource type. The goal of presented work is to build a workflow which will access and eventually store CAD data into IDAM. CAD data in IDAM has a single source, i.e. it is stored in single location as CATIA files. A first step is to build data resource which will include a metadata and a catalogue design. This step includes categorising the resources and record them to IDAM. The data resource will provide CAD data in STEP format for two workflows. The first workflow is the engineering one in which more complex 3D models are required in STEP format to be read by COMSOL and ANSYS and perform, for example, electro-magnetic nume rical simulations. The second workflow is the scientific one, in which 2D axi-symmetric section-cuts are performed on different levels of detail. The section cut is performed for specific angle from the horizontal axis. These section-cuts are then converted/exported by a python plug-into VTK/XML and used by a physics code EFIT++. Attention is given to definition of a structure in the STEP file in order to locate different components needed for the code. Generally, it is better to have one file to ensure the provenance. With such zero-copy approach, type movement could be achieved for better efficiency. The second step is to develop an IDAM server plugin to get/put metadata for CAD data objects into an object store. This will provide to serve the data through the plug-in. The storage includes collection and cataloguing of metadata during the CAD data handling. This ensures CAD data provenance tracking and capture together with other objects available in IDAM.

905

Effect of multilayer insulation on thermal loading in DEMO systems

Ingrid Vavtar, Martin Draksler, Boštjan Končar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
ingrid.vavtar.93@gmail.com

This paper discusses the radiative heat exchange amongst the major components of the DEMO tokamak. Since additional multilayer insulation at the warm side of thermal shields can substantially reduce the heat load to the magnets and thermal shields themselves, different shielding configurations, including those with passive multilayer insulation were investigated. Numerical analysis shows that regardless of the additional insulation layers being used, the excessively high thermal loads on the magnet system can be avoided only if the magnets' thermal shielding is actively cooled. Discussion is supported by a simplified theoretical model, which is in good agreement with the numerical predictions.
Based on the CAD model of the baseline DEMO design a 1/18th section of a tokamak geometry was created using the ANSYS DesignModeler software and used for a thermal radiation analyses with the Finite Element code ABAQUS. Careful meshing and input model development was required to reduce the global numerical error of radiation analysis to acceptable level. Based on the numerical analysis, the thermal loads on individual component as well as the amount of energy being exchanged in the system have been estimated. This allowed us to investigate the effect of the multilayer insulation, and to estimate the required power for active cooling of thermal shields.

Nuclear Fusion and Plasma Technology

906

Synthesis of W-based composite as a plasma facing material

Andreja Šestan, Matej Kocen, Janez Zavašnik, Saša Novak, Petra Jenuš, Miran Čeh
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
andreja.sestan@ijs.si

In previous fusion experiments (TEXTOR, ASDEX Upgrade and JET- Joint European Torus) tungsten has been used in plasma-facing components due to its acceptable thermo-physical properties. Compared to previous fusion reactors, materials incorporated in DEMO divertor will have to withstand even more extreme conditions. Despite tungsten's favorable properties, there are also several disadvantages that we will try to overcome, especially in terms of mechanical properties at high temperatures.
It has been suggested that by reinforcement of W-matrix with carbides (TiC, ZrC and HfC) refractory particles or oxide particles (Y2O3 and La2O3), tungsten mechanical properties can be improved [1]. As an alternative, we use W2C and WC particles to reinforce W-matrix. For this purpose, tungsten matrix is being reinforced with 10 vol. % of carbon precursor. The starting mixtures were prepared following the same procedure: tungsten powder was mixed with phenol formaldehyde resin. Homogeneous powder mixture was first uniaxial and then isostatically pressed. Two different temperature regimes in high-temperature furnace were used in order to obtain high density materials with homogeneous distribution of W2C particles in W-matrix. As-sintered samples were analyzed in terms of phase composition (XRD), microstructure (SEM) and mechanical properties (room temperature flexural strength).
The introduction of the two step sintering regime enabled the phenol formaldehyde resin to completely degrade into carbon, which was not achieved with the first sintering program. In our future work, the sintering process will be further

optimized in order to obtain composite with a high density (up to 95 % of theoretical density).

Reference:

1. Pintsuk, G., 4.17 - Tungsten as a Plasma-Facing Material A2 - Konings, Rudy J.M, in Comprehensive Nuclear Materials. 2012, Elsevier: Oxford. p. 551-581.

Nuclear Fusion and Plasma Technology

907

Tunnel probe measurements in a low-temperature magnetized plasma

Jernej Kovačič[1], James Paul Gunn[2], Tomaž Gyergyek[3]

[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia

[2]CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, F-13108 Saint-Paul-Lez-Durance, France

[3]University of Ljubljana, Faculty of Electrical Engineering, Tržaška 25, 1000 Ljubljana, Slovenia

jernej.kovacic@ijs.si

The tunnel probe [1] is a relatively new type of electrostatic probe which is suitable for use in tokamaks. Unlike the Langmuir probe, which is most commonly used in tokamaks, it has a concave geometry and works in a way like an inside-out Langmuir probe. The main advantage of thedesign is that the measured current-voltage characteristics do not suffer from the geometrical effect of sheath expansion. Therefore, the ion branch of the characteristic is properly saturated and can be used for plasma parameter evaluation without the error of collection area enlargement. The tunnel probe is divided into two biased collection areas, the back plate and the tunnel. The characteristic of the probe was extensively modelled using dedicated particle-in-cell simulation in order to make a perfect calibration of the probe measurements for different plasmas. Now, only by measuring the ratio of ion saturation currents from both collectors, ion current density and electron temperature can be measured simultaneously in a fast way.

Up until now the tunnel probe has only been used in tokamaks. We have now installed one such probe in a low-temperature plasma of the Linear Magnetized Plasma Device (LMPD) on Jožef Stefan Institute. Our goal was to study the behaviour of the probe in various plasma conditions to try and expand the range of the possible use of the probe. Since LMPD is a versatile machine, magnetic field intensity and angle, plasma density, electron energy distribution function, ion beams etc. can be altered or added, so we performed a number of measurements in different conditions. We were especially interested in the electron branch of the characteristic, since the discrepancy between the measured and the simulated results for that part for tokamak plasmas was significant. We also focused on the effect of the magnetic field inclination on the electron collection inside the tunnel. In this paper we shall present the results of the measurements on the LMPD.

[1] J. P. Gunn, R. Dejarnac, J. Stöckel, "Simultaneous DC measurements of ion current density and electron temperature using a tunnel probe", Journal of Physics: Conference Series 700 (2016), 012018.

908

Deuterium atom loading of self-damaged tungsten at different sample temperatures

Anže Založnik[1], Sabina Markelj[1], Thomas Schwarz-Selinger[2], Klaus Schmid[2]
[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]Max-Planck-Institut für Plasmaphysik (IPP), Boltzmannstr. 2, D-85748 Garching b. München, Germany
anze.zaloznik@ijs.si

During the operation of a fusion device, 14.1 MeV neutrons will be created. These high energy neutrons will produce defects in the bulk of the material of plasma facing components, degrading the favorable properties of the material and enhancing the fuel retention. In order to mimic the neutron-damaged tungsten, self-damaged samples i.e. implanted with high energy W ions, are used in experiments.

The edge plasma in a fusion device will consist of ions as well as of neutral atoms and molecules. The study of neutral particle interaction with plasma facing components is important in order to predict and understand the fuel dynamics in the divertor and to estimate the contribution of neutral particles to the overall retention of the fuel in the wall of a fusion device. In contrast to ions, neutral particles cannot penetrate directly into the bulk of the material, but are rather adsorbed on the surface. From the surface they can desorb back to the vacuum or they can diffuse in the bulk, where they contribute to the overall retention. In order to understand the mechanism of surface to bulk migration, a series of dedicated experiments was performed with self-damaged tungsten.

Polycrystalline tungsten samples damaged by 20 MeV W ions up to 0.25 dpaKP at maximum peak damage were exposed to deuterium atom beam at the sample temperature of 450 K, 500 K, 550 K and 600 K for the same atom fluence. Time evolution of deuterium depth profile was followed in-situ and in real time during the exposure by Nuclear Reaction Analysis (NRA) technique. Deuterium populates only 20 % of the damaged layer at 450 K, whereas the whole damaged layer is saturated in the case of 600 K. Namely, the trap population is slower for lower temperature. After the exposure a thermodesorption spectroscopy (TDS) was performed in order to obtain information about the defect concentration in the sample and desorption energy of deuterium atoms.

Experimental results were modeled in order to obtain information about the adsorption site types and the height of the potential barrier for diffusion from the surface to the bulk. The TESSim code [1] for deuterium trapping and bulk diffusion was upgraded by a surface model and used for deuterium depth profile calculation. The modeling results were fitted to the experimental data and modeling parameters were determined.

We found a nice agreement between the modeling and the experimental results. The determined values of adsorption energy were found to agree with the values reported in literature. The heat of solution for tungsten, calculated from the height of the barrier for surface to bulk diffusion, was 0.335 eV. This is approximately three times lower compared to the value of 1.04 eV, reported in literature. Modeling with the literature value for the barrier height resulted in a huge discrepancy between experimental data and modeling results, indicating the need for the low value of the barrier height or a temperature dependence of some other modeling parameters, which were considered constant in the current model.

[1] K. Schmid, V. Rieger and A. Manhard, J. Nucl. Mater. 426 (2012), 247

909

Thermal loading of divertor cassette during maintenance conditions

Luka Klobučar, Boštjan Končar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
klobucar.luk@gmail.com

The demonstration fusion power plant DEMO is planned to be the last major step before the commercial fusion power plant. DEMO tokamak is composed of several systems operating at very different temperatures, either extremely high (divertor) or extremely low (superconducting magnets). During the reactor operation the divertor is subjected to the incident heat flux of removed plasma particles with values above 10 MW/m3. Such heat loads may eventually cause severe damaging and consequently the need for regular replacement of divertor cassette that is envisaged at two-year cycle. The cassette under replacement is unplugged from the cooling pipes, while the remaining cassette and the blanket are still actively cooled. Because of lack of internal cooling the detached cassette is heated up due to the decay heat in activated materials.

This study aims to evaluate the heat load on the divertor cassette during maintenance conditions taking into account the decay heat immediately after the reactor shutdown and one month after the shutdown. The possibility for external cooling of the divertor cassette under replacement will be investigated as well.

The thermal model of the DEMO reactor vessel will be developed that includes actively cooled vacuum vessel, blankets and divertor cassettes as well as the detached divertor cassette subjected to the decay heat and thermal radiation of neighboring components. The 3D thermal analysis will be performed with the ANSYS CFX tool and will be based on the most recent DEMO tokamak design with 54 divertor cassettes. The geometry will be prepared in ANSYS Design Modeler, while ICEM CFD will be used for numerical meshing. Before carrying out the real case simulations, the code validation study will be performed on the simplified analytical benchmark. The benchmark geometry will be resemble the 3D model of tokamak as far as currently possible but will be simple enough to allow analytical solutions. Several cases with different boundary conditions (passive thermal radiation, imposed temperature conditions…) on divertor cassette surfaces will be analyzed and special attention will be paid to the error assessment.

The main goal of the study is to evaluate the temperature distribution in the detached cassette (mainly maximum temperature in the cassette body and maximum temperature on the plasma-facing surface) under different heat load and external cooling conditions in order to define the maintenance criteria.

The first study of deuterium retention in tungsten simultaneously damaged by high energy W ions and loaded by D

Sabina Markelj[1], Anže Založnik[1], Thomas Schwarz-Selinger[2], Mitja Kelemen[1], Primož Vavpetič[1], Primož Pelicon[1], Etienne Hodille[3], Christian Grisolia[3]
[1]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
[2]Max-Planck-Institut für Plasmaphysik (IPP), Boltzmannstr. 2, D-85748 Garching b. München, Germany
[3]CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, F-13108 Saint-Paul-Lez-Durance, France
sabina.markelj@ijs.si

Tungsten or advanced tungsten alloys are considered to be the most suitable material for plasma-facing components in future fusion reactors such as DEMO. In these nuclear devices tritium retention in neutron damaged tungsten will become more significant issue. In order to study the influence of material irradiation by neutrons on fuel retention, high energy ions are used [1]. It was shown that fuel retention in self-ion damaged tungsten is strongly increased as compared to undamaged tungsten [eg. 2]. Till now all retention studies are performed by sequential high energy ion damaging and subsequent loading of the material with hydrogen isotopes. However, in a real fusion reactor both implantation of energetic hydrogen ions and neutrals as well as damage creation by the neutron irradiation will take place at the same time. The consequences for retention are unknown. It is well known that in some metals impurities such as hydrogen, change the behavior of defect creation and recovery, e.g. on vacancy migration during recovery stage [3].
In this contribution we present the first experimental results on simultaneous D atom beam loading and defect creation by high energy self-ion implantation in tungsten. In order to have a good database to compare with, we also studied deuterium retention in self-damaged tungsten by the different procedures of tungsten damaging by tungsten ions and sequential deuterium loading by neutral D atoms. To make one step further towards more realistic situation we have performed first study of simultaneous tungsten irradiation by 10.8 MeV W ions and D atom loading, atom flux of 4.5x1018 D/m2s, at five different temperatures from 450 K to 1000 K for 4 hours yielding a maximum 0.5 dpa damage dose. After the damaging and loading D depth profiles were measured by NRA using D(3He,p)4He nuclear reaction. For the 450 K case the atoms hardly penetrated in depth whereas in the case of 800 K -1000 K the atoms did diffuse through the damaging area in 4 h due to the faster diffusion. In order to determine how many traps were actually created in the material the samples were after simultaneous damaging & loading and NRA analysis exposed to D atoms for additional 19 h at 600 K, fluence 3.7x1023 D/m2. As expected the highest concentration was obtained for the 450 K case, decreasing with damaging temperature. The results were compared to different sequential damaging/exposure experiments. Synergistic effects were observed, namely, higher maximum D concentrations were found in the case of simultaneous damaging and exposure as compared to damaging at elevated temperatures without offering D. Therefore part of the defects that would annihilate at high temperatures does not due to the presence of solute deuterium atoms in the bulk that stabilize the defects. However, the deuterium retention is still lower as compared to sequential damaging at room temperature and defect annealing.

[1] W.R. Wampler, R.P. Doerner, Nucl. Fusion 49 (2009) 115023.
[2] O.V. Ogorodnikova, et al., J. Nucl. Mater. 451 (2014) 379
[3] H. Schultz, Mater. Sci. and Eng. A141 (1991) 149.
[4]A. Založnik, et al., Phys. Scr. T167 (2015) 014031.

911

Micro-NRA and micro-3HIXE with 3He microbeam on samples exposed in ASDEX Upgrade and pilot-PSI machines

Mitja Kelemen
Institut Jožef Stefan, Jamova cesta 39, 1000 Ljubljana, Slovenia
mitja.kelemen@ijs.si

Tungsten or advanced tungsten alloys are shown as promising materials for high heat flux plasma facing components in tokamak fusion reactors. During plasma operation of such device, the wall is subjected to severe physical conditions, which lead to erosion, deposition, fuel retention and lattice damage of the wall. To obtain a depth profile of retained fuel, i.e. deuterium, for plasma experiments nuclear reaction analysis (NRA) is often used via the D(3He,p) α reaction, making use of a resonance like cross section at the energy of 630 keV. A sequence of measurements at several beam energies is used to deduce the D depth profile [1].

In the case where the information on lateral distribution of deuterium is sought for, equivalent NRA method is applied at Jožef Stefan Institute (JSI) with a focused 3He beam [2]. Negative 3He ion beam is formed in the combination of duoplasmatron ion source and Li- change exchange canal. A setup for 3He and 4He gas mixing was built in the duoplasmatron housing to spare precious 3He gas. The 3He2+ ions are accelerated with tandem accelerator to energy of 3.3 MeV. Under such conditions, we are able to form a 3He beam with diameter of 10 µm and ion current of 300 pA. During the measurements, in total four detectors are used simultaneously: a thick-depleted implanted silicon detector for NRA, a RBS PIPS detector, a HPGe X-ray detector for detection of particle induced X-ray emission (3HIXE) and a PIPS detector for beam dose normalization that detects ions scattered from the beam chopper. By scanning the focused 3He beam over an area of 2.2X2.2 mm2 a good lateral resolution is obtained, providing information on the lateral distribution of deuterium and other elements in the sample.

In the presented study we analyzed two samples, 10-20 nm W deposited on nominal and smooth graphite substrate, that were exposed in ASDEX Upgrade machine. The purpose was to study the effect of surface roughness on the net erosion/deposition patterns of tungsten as well as compare the deposition of different impurities (carbon, nitrogen, boron) and deuterium on them [3]. In addition, a sample exposed to D plasma in the Pilot-PSI linear machine, was analysed. The sample had a 1.5 µm W+Y coating on Mo substrate. The sample was analysed also by Laser-induced breakdown spectroscopy (LIBS) in order to characterize and test the performances of this technique. In addition the elemental inventory with secondary ion mass spectrometry was measured and lateral profile was created. The results obtained by the micro beam analysis will be presented and discussed. Good agreement between the elemental distributions obtained by NRA, 3HIXE was obtained when comparing to LIBS and SIMS measurements. With these results we provide a new powerful analytical tool for elemental inventory of plasma facing materials.

References:
[1] Markelj et al. J. Nucl. Mater. 469 (2016) 133
[2] Pelicon et al, Nucl. Instr. Meth. B 269 (2011), 2317.
[3] Hakola et al. Phys. Scr. T167 (2016) 014026

912

Production of prompt and delayed gamma rays in fusion reactors

Dijana Makivič, Igor Lengar
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
dijana.mkv@gmail.com

Neutrons with an energy of 2.5 MeV originating from D-D plasma and energy of 14.1 MeV from D-T plasma, propagate through structural materials of a fusion reactor and collide with nuclei in the material. Prompt and delayed gamma rays are created in the materials. Measurement of the gamma ray spectra from the plasma is an important diagnostic tool for determining plasma properties such as identification of fast ion species, their tail temperature and relative concentration. The measured spectra is however disturbed by gamma rays from structural materials. It is important to evaluate the gamma rays which originate from the neutron activation of materials, in order that correction to the measured gamma ray spectra from the plasma can be made. Evaluation of delayed gamma rays is important for determination of dose rates and delayed heat after the shut down and for ensuring safe maintenance of fusion reactors.
Calculations of the production of prompt gamma rays in a fusion reactor were made using the Monte Carlo N-Particle Transport Code (MCNP). The MCNP6 code is currently capable to compute the creation of new nuclides from nuclear reactions with incident particles, but does not create new nuclides as the result of radioactive decay. In addition the production of delayed gamma rays from newly created unstable nuclides can be calculated within different time bins. The concentration of transmuted nuclides and delayed gamma ray spectra for different time bins were calculated with the MCNP6 code. Calculations were made for several materials in a simple geometry. The same calculation were made using the FISPACT inventory code, with the capability to calculate the activation and transmutation induced by neutrons, without performing the transport of particles. The comparison was made between results obtained from MCNP6 and from the FISPACT code in order to make the evaluation of MCNP6 new features and capabilities for production of new nuclides and delayed gamma ray.

Nuclear Fusion and Plasma Technology

913

Deuterium Removal from Self-ion Irradiated Tungsten by Annealing in Vacuum and Isotopic Exchange

Olga Ogorodnikova[1], Sabina Markelj[2], V. V. Efimov[1], Yu. M. Gasparyan[1]
[1]Moscow Engineering Physics Institute National Research Nuclear University, "MEPhI", Kashirskoye shosse 31, 115409 Moscow, Russian Federation
[2]Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
olga@plasma.mephi.ru

Tungsten (W) is a material for ITER divertor and primary candidate for plasma-facing material for DEMO. Materials for fusion and fission suffer from high dose neutron irradiation at high temperatures. The tritium (T) inventory issue is one of safety and nuclear licensing as well as cost; in ITER and in any successor facility there will be strict limits on the amount of tritium that may be trapped in the wall material. Consequently, the removal of hydrogen isotopes from radiation damage in tungsten is important from point of view of tritium safety of fusion reactors. In this work, Wsamples were first pre-irradiated with self-ions to generate radiation damage and then exposed to deuterium (D) plasma at 470 K. To study the removal of hydrogen isotopes from radiation-induced defects in tungsten (W) by annealing, the samples were annealed in vacuum at temperatures of 600, 700 and 800 K for 2 hours.Modification of the deuterium depth profile in self-ion irradiated tungsten after annealing was measured by nuclear reaction analysis (NRA) using 3He+ as the analysing beam. Kinetics of the D release from radiation-induced defects of specimens after different post-annealing treatments was studied by thermal desorption spectroscopy (TDS). It is shown that D removal from radiation-induced defects in W by annealing in a vacuum is more efficient at lower displacement per atoms (dpa). At 0.5 dpa, the D concentration at radiation-induced defects decreases by factors of ~2 and 3 at annealing temperatures of 600 and 700 K, respectively, and by a factor of ~18 at 800 K. In spite of efficient reduction of D at radiation defects at 800 K, the D concentration at radiation defects is still about two orders of magnitude higher than that at intrinsic defects in undamaged material.

TheD removal by the annealing in vacuum was compared with the D removal by the isotopic exchange using hydrogen (H) atomic beam [1].The annealing in a vacuum at 600 K for 20 hours of samples pre-exposed to atomic D beam at 600 K reduces the D concentration at radiation damage only by a factor of about 1.5. The efficiency of isotopic exchange at the same temperature is higher: the D concentration at a peak damagedecreases in3 times for 21 hours of the exposure to H atomic beam.In the present work, we show analytically that the efficiency of isotopic exchange increases with increasing the sample temperature, incident ion/atomic flux and incident H energy that consists with the experimental data.

[1] S. Markelj et al., J. Nucl. Mater. 469 (2016) 133

1001

MetroERM - Metrology for Radiological Early Warning Networks in Europe

Denis Glavič Cindro, Toni Petrovič, Matjaž Vencelj, Benjamin Zorko
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
denis.cindro@ijs.si

In an event of a major radiological emergency, the early and reliable knowledge of radioactivity concentrations in air, and subsequently the assessment of contamination levels of farmland and of dose rate levels in urban areas are of key importance in organizing sound countermeasures for the protection of the general public from the dangers arising both from direct external radiation and from intake of radioactivity by ingestion or inhalation of contaminated food or air.

Therefore in 2014, a 3-year EMRP joint research project Metrology for radiological early warning networks in Europe (MetroERM) has been launched. The aim of this project is to develop methods for the harmonization of reported values on both dose rate and airborne radioactivity concentrations so that data related to the same trans-boundary event measured by different networks using different detectors are directly comparable. This will allow consistent data collation and evaluation and consecutively reliable conclusions could be drawn by the responsible authorities. In addition, within this project the development of new measurement techniques based on novel spectrometry systems with state of the art detection materials, such as LaBr3, CdZnTe or CeBr3 is in progress with the aim to allow both the calculation of dose rates and the calculation of contamination levels including nuclide-specific information at the same time.

Another intention of this project is to improve the capacity of the early warning networks by the development of new methods and systems for rapid ad-hoc radioactive air concentration measurements to efficiently supplement global early warning data with accurate information on airborne radionuclide content. With the development of new modular air sampling systems which can be easily transported to locations for the detection of airborne radioactive particulate, the information content provided by early warning networks in real time will be considerably increased.

At JSI a novel portable aerosol sampling device which provides real time airborne radioactive particulate monitoring was developed. It incorporates a 1 inch CeBr3 scintillation detector with ~4% FWHM energy resolution at 662 keV positioned centrally within a concertinaed filter assembly. An improved air pump with stable high flow rates, up to 200 m3/h, enables low level airborne radioactivity detection. To perform gamma spectrometry, a fully digital signal processing unit with a 4k-channel MCA was developed in-house. The temperature drift of the detection system is compensated on the software side by a microcontroller-based system connected to a digital thermometer that is in good thermal contact with the detector. The same microcontroller unit is used to handle the user interface, via a 5 inch color touch screen, and handles all inputs/outputs (I/O), including 3G network communications. It enables a prompt and continuous online detection and data evaluation from remote stations, as well as remote control of the unit settings and functions. The system is incorporated in a heavy-duty portable case which can be easily transported to different measurement locations.

The project MetroERM and the JSI contribution within this project will be presented and discussed.

1002

New Ceramic Waste Forms for High Level Radioactive Wastes

Neslihan Yanikömer[1], Sinan Asal[2], Sevilay Haciyakupoglu[2], Sema Erentürk[2]
[1]Istanbul Gelisim University, Faculty of Engineering and Architecture , 34315, Avcilar, Istanbul, Turkey
[2]Istanbul Technical University, Energy Institute, 34469 Maslak, Istanbul, Turkey
nyanikomer@gelisim.edu.tr

High level radioactive wastes can be stored in geological repositories after immobilizing in glass, ceramics, glass-ceramics and glass composite materials. Although waste volumes can be reduced and products having high chemical and mechanical durabilities can be obtained with the waste vitrification method, the radioactive waste capacity in glass matrix is limited between 20-35% (wt). Later, the waste capacity of the composites increased to 50-70% (wt) with the development of ceramic materials and at the same time materials having high resistance can also be obtained.

Purpose of this study is to prepare of different ceramic matrices having high chemical and mechanical durability, high waste capacity, a processing temperature as low as possible and having shielding feature against radiation from the radioactive waste in their matrices. Immobilization of some important fission products (137Cs and 90Sr) was performed in these ceramics prepared in proper composition. Then, the suitability of the main matrices and matrices containing wastes to the long-term underground storage conditions was investigated with detailed testing.

Keywords: Nuclear waste; Ceramics; Cs-137; Sr-90; Chemical durability; Leaching

Radiation and Environment Protection

1003

Dual track approach to strategy and planning for high level waste and spent fuel deep geological disposal

Tomaž Žagar, Leon Kegel, Matej Rupret
ARAO – Agencija za radioaktivne odpadke, Celovška cesta 182, 1000 Ljubljana, Slovenia
tomaz.zagar@gov.si

The paper will give overview of Slovenian national plan and strategy for managing radioactive waste and spent fuel with the focus on the high level waste and spent fuel management via dual track approach to deep geological disposal as the final solution for spent fuel and high level waste. Dual track approach is a novel and modern approach to treat in parallel international/regional deep geological repository development and national disposal program development.

Slovenia has a very small nuclear program: it owns one nuclear power plant in co-ownership with Croatia in 50:50 share located in Krško, Slovenia (Krško NPP). In addition to operating nuclear power plan there is also one research reactor (TRIGA) and central interim storage facility for radioactive waste from small producers, both near capital Ljubljana.

International and national regulatory and legal frameworks require a national programme for managing radioactive waste and spent fuel. The main goal of these programmes is to ensure safe and efficient management of radioactive waste and spent fuel. Experience shows that the route to an operational deep geological disposal facility is long and burdened with uncertainties, even for large nuclear programs. For countries with small or very small nuclear programs the financial and human resources required for the construction and operation of a geological disposal facility are significant, this is why idea

of regional and international cooperation regarding radioactive waste management is not new and has its roots in the previous century. However, due to uncertainties over implementation of regional/multinational repository, the national repository option must be kept open in national programme. Both plans can be implemented in parallel in national programmes in so called dual track approach.

Radiation and Environment Protection

1004

Assessment of Spent Fuel Activity in Dose Projection Software

Matic Pirc[1], Borut Breznik[1], Primož Mlakar[2]

[1]Nuklearna elektrarna Krško, Vrbina 12, 8270 Krško, Slovenia

[2]MEIS storitve za okolje d.o.o., Mali Vrh pri Šmarju 78, 1293 Šmarje-Sap, Slovenia

matic.pirc@nek.si

Dose projection software installed at Krško NPP is using an on-line calculation of radioactivity of the reactor core but until recently the software had no possibility to provide a quick dose assessment in case of overheated fuel accident in the spent fuel pit (SFP). Besides the long term inventory in the SFP, there is also an unloaded core during each refueling outage and about half of the core remains there after the outage.

The software should be able to provide a continuous activity calculation based on the operator input regarding inventory changes in the SFP. Some possibilities were verified to get a quick input based on a simplified approach using pre-calculated results of the Origen computer code.

Three different options regarding spent fuel inventory and occurrence of the accident were taken into account – before the unloading of the reactor core, during the fuel unloading/loading and after the fuel loading, considering also an annual recalculation by the Origen.

Radionuclides of the main interest are noble gases and other volatile elements at higher temperatures such as radioactive cesium and iodine which could be released into the air. Due to overall uncertainty related to dose calculation, the on-line calculation of SFP source term could be helpful in case of emergency calculations.

Radiation and Environment Protection

1005

Radiological consequences of potential disintegration of U tailings pile at the former Žirovski Vrh uranium mine, Slovenia

Tea Bilić-Zabric

INKO svetovanje, d.o.o., Kolezijska 5a, 1000 Ljubljana, Slovenia

inko@siol.net

The uranium mining complex at Žirovski Vrh was in operation during 1985-1990 and produced 455 tonnes of yellow cake. About 610.000 tonnes of technological waste were deposited on the elevated slope located nearby the mine complex. After extremely heavy and long lasting rain in autumn 1990, the upper part of the slope (together with tailings pile) started to slide with the progress of 2-3 cm per month. In spite of performed technical measures a land-sliding of the disposal site was not completely stopped and represents quite a real threat to the public and the environment.

Two research studies on the consequences of total disintegration of the tailings pile were elaborated on the initiative of the ministry responsible for environment. The preliminary geological study was dealing with the spread and distribution of rubble with radioactive waste material downstream the local valleys. In the subsequent radiological study the possible radiological consequences for the local population were worked out.

The paper presents firstly main results of the geological study on rubble deposition along two local streams for two simultaneous emergency events: a local earthquake and extreme precipitation with a 100 years return period and with a 1000 years return period. The results of this study – such as location and size of the areas of transported material, layer thickness and radioactivity of deposits - were used as input data for radiation exposure assessment. The evaluation of public exposure due to total disintegration of the disposal site was done for cases whether restoration takes place or if it does not. Dose assessment covered all important exposure pathways (inhalation, ingestion, external radiation) originating from the deposited radioactive material. Since the material would reach the settlements, i.e. houses, gardens, fields and grassland, the expected radiation exposure would be an order of magnitude higher as in the operational period of the U-mine. Whether restoration is not carried out the general dose limit for population of 1 mSv/y would likely be exceeded (estimated doses 1.3-4.5 mSv/y). Two main contributors to the public exposure would be the inhalation of radon with its short-lived progeny and external radiation (over 95 %). The ingestion of contaminated garden crops, milk, eggs, etc. would be of the order of 0,05 mSv/y and would not be of concern. On the other side, applied remediation measures would efficiently reduce public exposure down to the levels of 0.1-0.2 mSv/y. For a comparison, the current public exposure - after the completed restoration of the U-mining site - amounts 0.1 mSv/y or less.

Radiation and Environment Protection

1006

Ionization Smoke Detectors in Slovenia – Current Status and Future Challenges

Simona Sučić, Marko Kostanjevec, Tomaž Žagar
ARAO – Agencija za radioaktivne odpadke, Celovška cesta 182, 1000 Ljubljana, Slovenia
simona.sucic@arao.si

Ionization smoke detectors use an ionization chamber and a radionuclide such as americium-241 to detect smoke. In Slovenia, installation of ionization smoke detectors started at the beginning of the 1970's. Due to good technical performance, ionization smoke detectors were widely used and can be found in industrial, service and even in general public buildings like hospitals or schools. Significant numbers of ionization smoke detectors are still in use, however the last 10 years they are increasingly being replaced by new technologies. In practice, major reconstructions and upgrades of the existing fire alarm systems usually occur during the renovation of buildings and in that process old ionization smoke detectors are also being removed. According to the Slovenian legislation removed detectors are collected and transmitted to the ARAO - organization responsible for the public service of institutional radioactive waste management. Due to the large number of ionization smoke detectors, it is becoming a routine practice in their treatment that the device is dismantled and the associated radioactive source is recovered and conditioned for storage. The rest of non-radioactive materials (plastic, metal and electronic components) are treated and prepared for recycling. After such treatment the radioactive part of smoke detectors is in more appropriate form for storage and requires significantly less space in the storage facility. This paper presents current status and lessons learned related with the treatment of smoke detectors that were collected in recent years through the public service of institutional radioactive waste management in Slovenia. Also, future challenges and positive effects of performed actions related with the treatment of ionization smoke detectors in Slovenia are discussed.

1101

Public Opinion about Nuclear Energy – Year 2016 Poll

Radko Istenič, Igor Jenčič
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
radko.istenic@ijs.si

The Information Centre that was established within the Nuclear Training Centre at the Jožef Stefan Institute more than 20 years ago informs the visitors about nuclear power and nuclear technology, about radioactivity and about Krško Nuclear Power Plant.
Information activities are targeted mainly at schoolchildren from the 8th and 9th grade of elementary school with their teachers (in total close to 8000 per year). The visit consists of a live lecture about nuclear technology followed by the demonstration of radioactivity and a guided tour of a permanent exhibition.
The opinion trends are monitored since 1993 by polling about 1000 youngsters every year. The poll is conducted before the youngsters listen to the lecture or visit the exhibition in order to obtain their opinion based on the knowledge from everyday life. Trends over the last 23 years will be presented, summarized and commented.

Education, Public Relations and Regulatory Issues

1102

Energy for Children

Vesna Slapar Borišek
Jožef Stefan Institute, Jamova cesta 39, 1000 Ljubljana, Slovenia
vesna.slapar-borisek@ijs.si

At the Information Center that was established within the Nuclear Training Center at the Jožef Stefan Institute visitors are informed about nuclear technology, the basic properties of radioactivity and protection against ionising radiation.
Lectures and demonstrations about nuclear technology and radiation protection are intended primarily for youngsters in last couple of grades of primary school and secondary school students. Occasionally our visitors are pupils from lower grades of primary school and even kindergarten children. For these children lectures about power plant operation and nuclear technology are too demanding and should be therefore appropriately adjusted according to the level of their knowledge. For this purpose, we decided to prepare a new special lecture with demonstrations which is intended for children of all ages. With this lecture and demonstrations we want children to understand the concept of energy and energy conversion. In addition we want to show to the children the technology development from Heron's Aeolipile, to the steam engine and the turbine and its use in power plants.

1103

European Decommissioning Academy (EDA) – 2nd run

Vladimír Slugeň, Martin Hornáček, Róbert Hinca, Filip Osuský
Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Institute of Nuclear and Physical Engineering, Ilkovičova 3, 812 19 Bratislava 1, Slovakia
martin.hornacek@stuba.sk

According to analyses presented at EC meeting focused on decommissioning organized at 11.9.2012 in Brussels, it was stated that at least 2000 new international experts for decommissioning will be needed in Europe up to 2025, which means about 35 per year. This growing decommissioning market creates a potential for new activities, with highly skilled jobs in an innovative field. A clear global positioning of the European Union is beneficial and will stimulate export of know-how to other countries, especially those having a large nuclear programme, and promote highest safety levels.
Having in mind the actual EHRO-N report from 2013 and 2014 focused on operation of nuclear facilities [1] and an assumption that the ratio between nuclear experts, nuclearized and nuclear aware people is comparable also for decommissioning, as well as the fact that the special study branch for decommissioning in the European countries almost does not exist, this European Decommissioning Academy (EDA) could be helpful in the overbridging this gap.
European Decommissioning Academy was created at the Slovak University of Technology in Bratislava Slovakia, based on discussion and expressed needs declared at above mentioned meetings and reports including ECED2013 conference.
The main goal of the Academy is from nuclearized experts (graduated at different technical universities and increased their nuclear knowledge and skills mostly via on-job training and often in the area of NPP operation) to create nuclear experts for decommissioning, which includes the lessons, practical exercises in our laboratories, on-site training prepared at Jaslovské Bohunice and Mochovce sites, Slovakia. Technical tour via most interesting European facilities in Swiss and Italy are part of the course as well [3].
The first run successfully passed 15 participants during June, 7 – 26, 2015. After the final exam, there was an option to continue in knowledge collection via participation at the 2nd Eastern and Central European Decommissioning (ECED) conference in Trnava (Slovakia) [4].
Based on the lessons learned during the first run of EDA and the feedback of the participants we now plan the 2nd run of the European Decommissioning Academy. The Academy is planned in June, 4 – 22, 2017 (including 3rd Eastern and Central European Decommissioning (ECED) conference in Trnava, Slovakia in June, 20 – 22, 2017).

Further information except the references can be also found at http://kome.snus.sk/inpe/ which is being actualised.

Acknowledgements
Authors thank to the Slovak Government grants VEGA 0204/2013 and to 001STU-2/2014-CEPVYJZ. We highly acknowledge also IAEA for support.

References

[1] European Nuclear Education Network, [online], available: <http://www.enen-assoc.org>.
[2] SLUGEN, V., HINCA, R.: European Academy of decommissioning: In XXXVI. Days of Radiation Protection Book of astracts, Poprad 2014, Slovak Republic. ISBN 978-80-89384-08-2, EAN 9788089384082
[3] SLUGEN, V.: European Decommissioning Academy (EDA): Ready to start. In International Journal for Nuclear Power, ISSN 1431-5254, 2015, vol. 60, no. 2, pp. 82-84.
[4] SLUGEN, V., HORNACEK, M.: European Decommissioning Academy – successful 1. run in June 2015. In: Proceedings of the Eastern and Central Europe Decommissioning ECED 2015, 23-25 June, 2015, Trnava, Slovak Republic. ISBN 978-80-971498-5-7.

Challenges in Education, Training and Knowledge Management

Sustainable development of the humankind in the future will, among others, require access to sufficient, environmentally acceptable and affordable energy sources. Development of abundant and affordable low carbon energy sources might well represent one of the most important and complex challenges that humankind will have to adequately solve within a few decades.

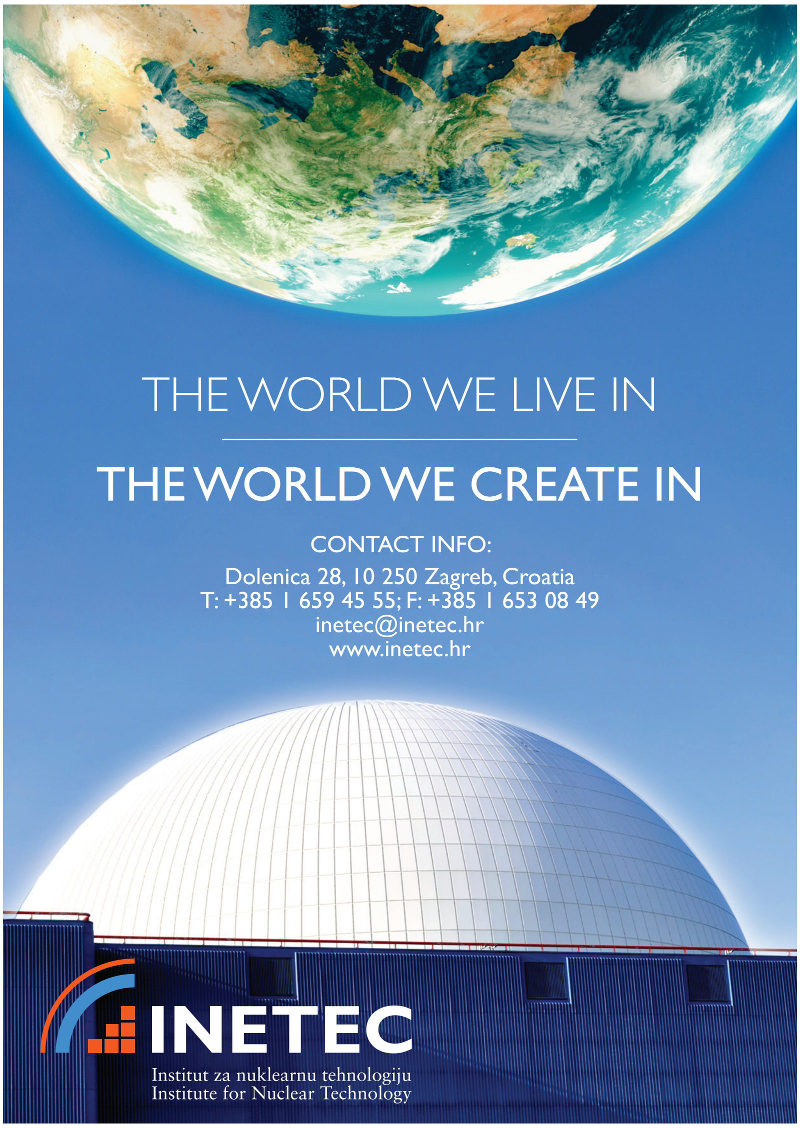
The sheer complexity of this challenge calls for new knowledge, excellently educated and motivated individuals and intensive knowledge management activities within all nuclear stakeholders.

Preliminary panelists:

- Prof. Michel Giot, Professor Emeritus, Université catholique de Louvain
- Dr. Franck Wastin, Head of Unit Knowledge for Nuclear Safety, Security and Safeguards, European Commision, DG Joint Research Centre
- Dr. Roger Garbil, Scientific officer, European Commission, DG RTD, Unit Nuclear fission
- Mr. Robert Stakenborghs, General Manager, ILD Evisive, Baton Rouge, LA, USA, Chair of Executive Commitee of the ASME Nuclear Engineering Division
- Dr. Asif Arastu, Unisont Engineering, San Francisco, CA; USA, Past Chair of Executive Commitee of the ASME Nuclear Engineering Division
- Mr. Clayton Smith, Director, Technical Services, Fluor, Greenvile, SC, USA, Member, ASME Commitee on Nuclear Codes
- Prof. Leon Cizelj, Head, Reactor Engineering Division, Jožef Stefan Institute, Slovenia, and President of the European Nuclear Education Network (ENEN)

Notes

Notes


THE WORLD WE LIVE IN
THE WORLD WE CREATE IN
CONTACT INFO:
Dolenica 28, 10 250 Zagreb, Croatia
T: +385 1 659 45 55; F: +385 1 653 08 49
inetec@inetec.hr
www.inetec.hr
INETEC
Institut za nuklearnu tehnologiju
Institute for Nuclear Technology


elmont
KRŠKO
Elmont d.o.o. Krško, Cesta krških žrtev 135e
8270 Krško, Slovenija
T: +386 7 49 12 500 / F: +386 7 49 12 521
E-naslov: info@elmont-kk.si
www.elmont-kk.si
SYSTEM CERTIFICATION
ISO 14001
SGS
Cert. No. CH02/0130
SYSTEM CERTIFICATION
OHSAS 18001
SGS
Cert. No. CH13/0758
SYSTEM CERTIFICATION
ISO 9001
SGS
Cert. No. CH97/0154
Po poti odličnosti

25th International Conference
Nuclear Energy for New Europe
www.nss.si/nene2016