8th Young Generation Nuclear Conference

BOOK OF ABSTRACTS

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Book of Abstracts of

8th Young Generation Nuclear Conference

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Foreword

The 8th Young Generation Nuclear Conference was the first young generation conference to be held virtually. Traditionally we have all gathered at the reactor centre in the beginning of March, but despite our best efforts, the situation this year has only enabled us to meet online. While the regular coffee breaks and related fruitful discussions will be missed, the online format enabled the presence of remote speakers and participants, which could otherwise not attend the conference.

In the times in which the networking is more important than ever, the conference again connected the young authors from different fields and enabled them to present their contributions to a broader audience. We hope that you found the whole meeting interesting and intriguing and that we meet again in person next year.

Organizing committee
Boštjan Zajec and Jan Kren

Acknowledgements

The organization of the conference would not be possible without the wide support. Thanks to Nuclear Society of Slovenia and Reactor Engineering Division for all the help with the organization of the conference. Special Thanks to our sponsors NEK and GEN ENERGIJA for making our conference possible.

Special thanks to Janez Kokalj, the president of Young Generation Network of Nuclear Society of Slovenia for advices. Thanks to Anže Pungerčič and previous organizers for providing a knowledge base to start the conference on. Thanks to Bojan Žefran for the technical support. Special thanks goes to the president of the Nuclear Society of Slovenia, dr. Tomaž Žagar, for the help with promotion and organization of the conference.

Last but not least, big thanks to all the authors and participants of the conference. With your presentations, opinions, questions and answers, you are the reason for the conference to be so intriguing.

Organizing committee

Our sponsors:

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Invited lectures

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Calculation of gamma and neutron dose field inside the JSI TRIGA Mark II reactor hall

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This presentation describes the development of a stochastic method to calculate gamma and neutron dose rates for the JSI TRIGA reactor, which can be applied to normal and emergency operations. Knowing the dose rates during normal operation is essential to keep workers safe and in the design of appropriate shielding for new experiments while knowing the dose rates during a postulated accident scenario is necessary for developing an emergency response plan.

A completely new MCNP model was designed containing the reactor core and the surrounding components within the concrete shield. Furthermore, the reactor platform, reactor hall, reactor basement and the control room were included in order to calculate the gamma and neutron dose fields within the radiation controlled area. Neutrons and prompt gamma rays were validated in the case where a beam tube was left open, and the reactor was at low power. In this way, neutron and gamma dose rate measurements could be taken around the beam port. The delayed gamma source was validated in the case where one of the irradiated fuel elements was placed in a transport cask, and the surrounding dose rates measured.

The good agreement between the calculated and measured results meant the model could be used to predict dose rates during normal operation with newly designed shielding for the beam tube no. 5. Before the shield was constructed, its performance was evaluated by the methodology developed in this thesis. Furthermore, an accident scenario involving the loss of water (LOWE) in the reactor pool and the spent fuel pool were analysed using the same methodology. The LOWE scenario is one of several design-based accidents scenarios to be considered when operating the JSI TRIGA reactor, that was analysed for the first time by the same method. The LOWE for the reactor pool was previously analysed using deterministic methods. Provided results will be used for the next revision of the Safety Analysis Report of the JSI TRIGA Mark II research reactor.

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Rosatom projects worldwide and technical characteristics of VVER technology

A. Ovcharenko, A. Renev

Rosatom is a unique vertically integrated enterprise that covers the complete nuclear fuel cycle from uranium mining throughout NPP construction and operation to back-end and decommissioning. Rosatom also develops its business and research in non-energy sectors such as research reactors, radiation technologies and nuclear medicine.

Being a global player with many projects in Russia and overseas, Rosatom keeps leading role in NPP project development and providing Russia and overseas partners with stable and clean electricity supply. Currently Rosatom holds approximately 70% of the global NPP construction market of overseas projects. Now Rosatom has with 35 power units at different stages of implementation in 12 countries. For the past 14 years, Rosatom has commissioned 15 NPP units both in Russia and abroad.

VVER-1200 reactor is Rosatom's safe and economically efficient flagship reactor technology. It is a time-tested and highly referential energy generating solution of generation III+, designed in strict compliance to post-Fukushima safety requirements. Generation III+ nuclear power plant with VVER-1200 technology combines successful experience in NPP operation with cutting-edge safety standards, while meeting the most stringent requirements.

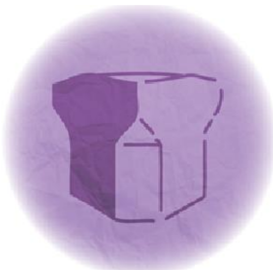
The main principles and technical specifications underpinning the VVER-1200 design are:

- maximum use of proven technologies;
- balanced combination of active and passive safety systems;
- 1500 reactor years of safe operation;
- 1200 MWe nominal output;
- 60+ years lifecycle;
- >90% availability factor.

Russian-designed VVER reactors have successfully undergone international stress testing. VVER reactors are operated in a number of the EU countries, including Finland, Czech Republic, Hungary, Slovakia and Bulgaria.

II

Research reactors and nuclear data



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Development of RAPID extension for TRIGA reactor 3D burnup calculations

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Determination of accurate 3D pin-wise fuel burnup in nuclear reactors is vital from the standpoint of fuel management, spent fuel storage safety and safeguards. In addition, the need for accurate and efficient burnup calculations has become more urgent for the simulation of advanced reactors and monitoring of spent fuel pools. To accomplish this, the Virginia Tech Transport Theory Group (VT^3G) has been working on advanced computational tools for accurate modeling and simulation of nuclear systems in real-time. One such capability is a novel methodology for performing 3D fuel burnup calculations, bRAPID, which utilizes the RAPID Code System. RAPID is based on the Multi-stage Response-function Transport (MRT) methodology [1] [1], that decouples a problem into independent stages that are then coupled in real-time via transfer functions/coefficients.

Recently, we initiated activities to benchmark the bRAPID methodology using the well characterized Jozef Stefan Institute's TRIGA Mark-II research reactor. Thus far, we have created a database including full operational history that allows for burnup validation possibilities in the form of measured excess reactivity[2].

Our talk will focus on the extension of the bRAPID algorithm for its application on the TRIGA reactor. In particular, we will focus on bRAPID's database pre-calculation procedure in which the Fission Matrix (FM) coefficients for different combination of reactor power and irradiation times are calculated. The FM coefficients are dependent on fuel burnup and change with U^{235} depletion, Pu^{239} production and reactor poison (xenon and samarium) formation. The evaluation such changes is crucial for the development of the bRAPID burnup methodology.

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[1] A. Haghighat, K. Royston, and W. Walters. "MRT methodologies for real-time simulation of nonproliferation and safeguards problems." Annals of Nuclear Energy, volume 87, pp. 61–67 (2016).

[2] A. Pungerčič, D. Čalič, L. Snoj, "On the Burnup of the JSI TRIGA MARK II Research Reactor Fuel", In review in Progress of Nuclear Energy, 2020

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First-Principles Study of ZrH

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Currently used thermal scattering nuclear data are based on old evaluations or evaluations which have been improved only slightly over the past few decades. For present needs old nuclear data are sufficient as current reactors typically operate at constant, full power. For the purpose of following the production of electricity from small modular reactors new sets of nuclear data will be needed, which in addition to the data themselves will have estimated uncertainties. Firstly, the goal was to get acquainted with the procedure and programs for the generation of thermal scattering nuclear data. Thermal scattering nuclear data for ZrH were generated.

With the advancement in computational power, state-of-the-art atomistic simulations based on first principles have become available for use in chemistry and physics. In these simulations, crystal lattice motion or molecular motion is simulated using algorithms from the field of computational physics, such as density functional theory and its derivatives. Density of states can be extracted from a simulated motion of an atomic or molecular system. Thermal nuclear data calculations are possible based on the extracted density of states. In density functional theory the total electronic force is the sum of the Coulomb repulsion between nuclei and the quantum mechanical electron potential, which itself consists of electron-electron interactions and electron-atom interactions. A fundamental assumption of density functional theory is that the total energy of the electronic system can be expressed as a function of the electronic density, and that the ground state of the system can be obtained by means of minimization of the function. In principle, success of density functional theory is due to the fact that there exists a bijective transformation between wave-functions of the constituent electrons and the full molecule electron density: the many-electron problem is reduced to a one-electron problem. To accurately simulate the behaviour of an atomic or molecular system, the interatomic (intermolecular) potential must be known so forces on constituent atoms (molecules) can be calculated. Using density functional theory capable computer codes, such as VASP [3], the system is modelled and relaxed to its ground state. Atomic positions are then perturbed, and the interatomic force constants are calculated. Once the force constants are obtained, they are transferred to another program, such as PHONON [4] or Phonopy [5],

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[3] Kresse, G., Hirsch C., https://www.vasp.at/, 2018.
[4] Parlinski, K., Phys. Rev. Lett. 78, 4063 (1997); K. Parlinski, Computer code phonon Cracow, 2012.
[5] Togo, A., Scr. Mater., 108, 1-5, 2015.

which performs lattice dynamics calculations in which the solutions to the dynamical matrix problem are sought. The solutions constitute the dispersion relation of the system, from which the density of states can be computed using a geometrical procedure. Once the density of states has been obtained, the scattering law is traditionally calculated using the LEAPR module of the NJOY [6] code package.

[6] *Macfarlane, R., The NJOY Nuclear Data Processing System, Version 2016, LA-UR-17-20093.*

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Modelling of OPEN100 Reactor with Serpent-2

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Several new nuclear power plants have been facing rising construction costs and delays due to overly complicated and proprietary designs. A possible solution is offered by the OPEN100 open-source PWR reactor design. The main idea is to have a smaller reactor with a longer fuel cycle where reactivity balance is established by the correct core design and the use of burnable absorbers. As the reactor is still in the design phase, lots of calculations are needed to analyse and optimize core design.

Publicly open CAD files, available on [7], were used to create a computational model in the Serpent-2 Monte Carlo neutron transport and depletion code. The created model is extremely important as it enables us to study the length of the fuel cycle, control drum reactivity worth, and other reactor physical parameters such as mean neutron lifetime, β_{eff} and neutron flux spectrum. The reactor's core consists of 24 fuel assemblies, each assembly having 264 fuel bundles, 24 guide tubes and one instrumentation tube. Reactivity control is made by 16 control drums surrounding the core. The final design and many reactor physical parameters have yet to be determined.

The analysis of the total worth of control drums was made with different core and drums designs. We have found that 120° angular B_4C absorber together with beryllium reflector results in control drums worth of 4000 pcm, which is comparable to the worth of control rods in standard PWR's [8].

The estimated electrical power of OPEN100 is 100 MW. This size fits 90 % of energy markets around the world. Smaller generating capacity means more flexible siting, the ability to use a standard supply chain, and access to a greater variety of capital sources. The initial burnup calculation was made considering the simple 2D geometry of one fuel assembly with periodical boundary conditions and the thermal power of 300 MW. We used different fuel enrichments (3 % - 5 %) and different burnable absorbers in the shape of cladding (ZrB_2) and fuel mixtures (CdO, Er_2O_3 and Gd_2O_3). The goal was to optimize the absorbers selection for the longest possible autonomous fuel cycle.

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[7] https://www.open-100.com/

[8] Anže Mihelčič, Anže Pungerčič and Luka Snoj, "Modelling of OPEN100 Reactor with Serpent-2", Proceedings of the 29th International Conference, Nuclear Energy for New Europe. 2020.

8th Young Generation Nuclear Conference, Brinje, 19th May 2021

Computation of k_{inf} from Monte-Carlo generated cross sections

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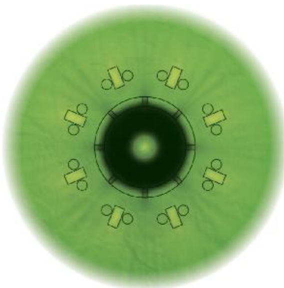
The recent increases in computational power have made stochastic codes an increasingly viable option for computation of homogenized cross sections. One option for comparing the homogenized cross sections between different codes is the integral quantity of k_{inf}, since it can be computed directly from group-averaged cross sections and does not consider neutron leakage. Since some Monte Carlo codes do not offer the possibility of computing the k_{inf} for a specific material directly, it can be computed through cross sections. Such approach is also useful for verification of computed cross section definitions – for example to check if the scattering matrix also includes reactions, which produce multiple neutrons.

The formula for k_{inf} can be derived starting from the two-group diffusion equation by neglecting the external source, assuming that all neutrons are born in the fast group and setting leakage/buckling to zero. The resulting k_{inf} matches the k_{eff} for systems without leakage as long as the cross-section definitions are used consistently. In this paper we are describing the comparison of k_{inf} computed directly from OpenMC and indirectly through the cross sections.

III

Fusion Reactor Physics



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Computation of Lithium Hydride Neutron and Gamma Ray Attenuation

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Fusion reactors present a challenging environment for spectroscopic measurements of gamma rays produced during fusion reactions inside the plasma. Therefore, gamma ray detectors are located far away from the plasma at the ends of collimated line of sights. Different absorbers are placed before the detector to reduce neutron background at detector position and, more importantly, to reduce neutron damage to the detector. The absorbers need to have a particular property, namely, they need to be strong neutron absorbers, but gamma ray absorption needs to be low. One of such materials is Lithium Hydride (LiH).

As the gamma detectors are located far from plasma, a unique computational model was made in MCNP to computationally determine the LiH attenuation factors for neutrons and gamma rays. The model consists of a Lanthanum Bromide detector and LiH shield of different thicknesses (66 cm and 93 cm, same thickness as used at JET). A separate analysis was performed for neutrons produced in deuterium-deuterium (DD) fusion reactions and deuterium-tritium (DT) fusion reactions and gamma rays of different energies.

The simulations have shown that for 2.5 MeV neutrons from the DD reactions the attenuation factor is around 5600 and 66000 for 66 cm and 93 cm thick LiH respectively. For 14 MeV neutrons from DT reactions, the attenuation factor is 530 and 6500 for LiH of a thickness of 66 cm and 93 cm respectively. In comparison, for low gamma ray energies (below 1 MeV), the LiH attenuation factor is above 100, while with higher energy the attenuation factor decreases and is below 10 for energies higher than 3 MeV for 93 cm thick LiH. The obtained neutron attenuation factors however differ from the factors found in the literature for 66 cm of LiH (900 for DD neutrons and 275 for DT neutrons). Thus, a new analysis is needed to determine the neutron attenuation factor based on computation with realistic MCNP neutron source and comparison with experimental measurements at tokamak JET.

Water activation experiment: PhD thesis topic overview

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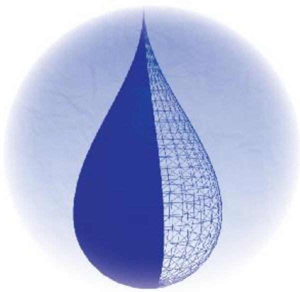
Water as a primary coolant is being used in most of today's fission reactors and will be used in ITER. It is also one of the most promising coolants for DEMO and other future fusion reactors. During cooling of the reactor blanket (fusion)/core (fission), the water is exposed to neutrons, produced in fusion/fission reactions, and gets activated, which leads to radioactive decay with releasing high-energy gamma rays and neutrons. Activation of cooling water predominantly consists of activation of oxygen isotopes via the $^{16}O(n,p)^{16}N$, $^{17}O(n,p)^{17}N$ and $^{18}O(n,\gamma)^{19}O$ reactions. Activated nitrogen and oxygen nuclides subsequently decay by emitting various decay products (gamma rays and neutrons) with different energies. As the cooling water loop extends outside of the primary biological shield and distributes the radioactivity to many critical components, additional protection for detectors, instrumentation, superconducting coils, and personnel must be considered.

Many computational analyses of the water activation process have been performed for ITER and DEMO; however, results are subject to enormous uncertainties and consequently poor quality due to lack of experimental nuclear data, inaccurate computational methodologies/codes and lack of experimental facilities for experimental validation of methodology.

This paper will present an overview of the proposed PhD thesis topic, which will focus on the development of the computational methodology and codes to model time and spatially dependent radiation sources used for the determination of dose fields due to the flow of the activated water in existing (JET) and future (ITER, DEMO) fusion devices, making significant contributions to the feasibility of future commercial fusion reactors. These methodologies will be experimentally validated by measurements on the proposed newly constructed irradiation facility at the JSI TRIGA reactor featuring a water activation loop. Furthermore, such a facility will enable various water activation-based experiments, which are essential to fill the knowledge gaps and to improve existent experimental nuclear data sets. It is the first time that the water activation problem will be tackled in a comprehensive and diverse way.

IV

Experimental Thermal-hydraulics



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PIV measurements of single- and two-phase turbulent flows in a vertical pipe

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Turbulent flows are common physical phenomena observed in natural world and in many industrial processes, including nuclear reactors. Understanding turbulent flows is therefore crucial to design better, safe and efficient thermal-hydraulic systems.

At THELMA laboratory of Reactor Engineering Division a Particle-Image Velocimetry (PIV) technique has been applied to study turbulent flow in a vertical pipe geometry. PIV is an optical method for flow visualization, which is frequently used to obtain non-intrusive velocity fields in gases and liquids. The fluid is seeded with tracer particles, which follow the flow dynamics. The fluid and particles are illuminated with strong and focused laser sheet and images of the flow are obtained with high-speed camera. Advanced numerical methods are then used to reconstruct the instantaneous flow fields as well as averaged first and higher-order flow statistics.

In this work, we present theoretical and technical background of the PIV measurements in single- and two-phase flows. We focus on current problems and good practices obtained while performing PIV measurements at THELMA laboratory. We also present current results for single-phase flow and future aspects of measurements in the two-phase (liquid-gas) regimes.

Abstract number: 402

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Measurement of turbulent flow over backward facing step with particle image velocimetry

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As certain geometries are interesting for studies on nuclear reactor cooling, backward facing step has been chosen as one of the key geometries. With the help of the Particle Image Velocimetry (PIV) technique water velocity fields in the flow over a backward facing step were measured [9]. The test section of length 1.2 meter was constructed as a hollow transparent channel with a step roughly in the middle. The experimental campaign described in the present work was performed in turbulent flow with Reynolds numbers around 7100. A single highspeed camera in combination with a pulse laser and LaVision software was used for a series of two-dimensional measurements of the velocity field at several cross-sections from two different perspectives. Subsequent experimental results presented in this work were compared with an accurate numerical simulation of a similar geometry with comparable Reynolds numbers.

[9] N. Kosanič, 2021, *Merjenje turbulentnega toka v kanalu s stopnico z metodo slikanja delcev*, Retrieved from: https://repozitorij.uni-lj.si/IzpisGradiva.php?id=124525

8th Young Generation Nuclear Conference, Brinje, 19th May 2021

Flow boiling in horizontal annulus – visualization and bubble size distributions

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Flow boiling is used as a heat transfer mechanism in many different systems where large surface heat flux is required. Notably, flow boiling also occurs in different types nuclear reactors, either as a part of process or during hypothetical accidents. An important aspect for design and safety of such systems is an accurate prediction of two-phase flow effects. Despite a long history of research, the underlying principles of boiling at higher heat fluxes and larger void fractions are still not well understood. For better understanding of basic phenomena, we conducted experiments in an annular test section, hydraulically following a geometry of a single reactor fuel rod. The experiment is a part of THELMA laboratory at IJS R4 department.

Subcooled flow boiling of the working fluid R245fa was investigated in a horizontal annulus and visualised with a high-speed camera at varied mass flow rates and constant inlet temperatures, covering Reynolds numbers from 1500 to 7300. As reported in our previous work[10], we observed qualitative changes in bubble behaviour with increasing the refrigerant mass flow rate. At lower mass flow rates large amounts of vapour gathers at the top, while at higher flows only smaller bubbles are present in the entire test section.

In this contribution we describe bubble behaviour by individual bubble counting and calculation of bubble size distributions. Changes in flow patterns translate to different bubble size distributions, which are bimodal at lower mass fluxes and become Rayleigh-like at higher mass fluxes. Despite noticeable change in bubble size distributions, we observed no large effect on the transferred heat flux.

---

[10] B. Zajec, B. Končar, M. Matkovič, L. Cizelj, *Eksperimentalno opazovanje konvektivnega vrenja v obročasti geometriji*, 7. Konferenca mladih jedrskih strokovnjakov, Brinje, 2020

V

Severe Accidents

8th Young Generation Nuclear Conference, Brinje, 19th May 2021

Overview report on the accident at the Fukushima Daiichi NPP (Summary of known facts - 10 years later)

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It has been 10 years since a major earthquake struck Japan which caused a tsunami that damaged, among other things, the Fukushima Daiichi NPP in Fukushima prefecture. The consequences of the nuclear accident are well known and greatly mitigated to this day. This report gathers conclusions and facts as an overview of the actual dimension of the nuclear accident and its impact on people, the environment and, consequently, the global nuclear field. Certain consequences of the earthquake and the devastating tsunami are presented, as well as the events in individual units of the power plant at the time of natural disaster. Next, the reasons for the accident and the radiological releases in Fukushima Prefecture are described. Following are the reasons and mechanisms that led to an accident of that magnitude and the lessons learned from the accident by the global professional public. Lessons range from organizational, regulatory to technological improvements at existing power plants and power plants in planning. After that, the ways to prevent additional environmental contamination and to repair existing damage are presented and as a conclusion, the current situation in the Fukushima Daiichi NPP and in the entire nuclear sector is described.

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Pool scrubbing simulations of SCRUPOS experiment

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During a hypothetical severe accident in a light water nuclear power plant, the reactor fuel could melt and there is a possibility, that some of the radioactive material could be released as particles to the surrounding area. The releases of the radioactive material can be reduced with the application of pool scrubbing, where the contaminated gases are filtered through a pool of liquid water. Since the gases in the scrubbing pool disperse into bubbles, the behavior of the particle removal from the bubbles is crucial for understanding pool scrubbing phenomena.

In this work, the multi-phase (liquid, gas, particles) simulations[11] performed using the open-source Computational Fluid Dynamics code OpenFoam's[12] solver reactingMultiphaseEulerFoam are presented. The simulated cases are SCRUPOS experiments[13]. In the end, the results were analyzed and the decontamination factor, which is the resulting measure of the scrubbing efficiency, was calculated.

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[11] H. Rusche, Computational fluid dynamics of dispersed two-phase flows at high phase fractions. 2003, Imperial College London (University of London).

[12] C. J. Greenshields, "Openfoam User Guide", OpenFOAM Foundation Ltd, version 3, 2015

[13] M. Turni, "Experimental study and modeling of a pool scrubbing system for aerosol removal", Ph.D. thesis , ETH Zurich, 2016

Abstract number: 503

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Assessment of premixed layer formation model on experimental results

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A hypothetical severe accident in a nuclear power plant can lead to significant core damage, including melting of the core. The hot melt in contact with the coolant water can result in a vapour explosion. Similar explosion phenomenon can be a threat also in some industrial processes, such as foundries and liquefied natural gas operations or in certain volcanic activity where water is present. In analyses of severe accidents in nuclear power plants, a fuel-coolant interaction was mostly addressed in a geometry of a melt jet poured into a coolant pool. Based on some experimental and analytical work from the past a geometry with a continuous layer of melt under a layer of water, called stratified configuration, was believed to be incapable of producing energetic fuel-coolant interaction. However, the results from recent experiments performed at the PULiMS and SES facilities (KTH, Sweden) with corium simulants materials contradict this hypothesis. In some of the tests, a premixing layer of ejected melt drops in water was clearly visible and was followed by strong spontaneous vapour explosions.

The purpose of our research was to improve the knowledge, understanding and modelling of the fuel-coolant interaction phenomena in the stratified configuration. A model for the premixed layer formation was developed. The model is based on the bubble formation, growth and collapse mechanism, which was visually observed in the experiments and suggested as the possible driving mechanism also from the authors of the experiments.

The developed model was implemented into the MC3D code (IRSN, France) as a patch and validated against the experimental results of PULiMS E6 and SES S1 experimental tests. The analysis of the simulated premixed layer formation demonstrates the model capability to describe the premixed layer formation in agreement with the experimental data.

Conference Invitation

Young Generation Network of
Nuclear Society of Slovenia

cordially invites you

to attend the

8th Young Generation Nuclear Conference

At Reactor centre, Jožef Stefan Institute, May 19, 2021.

The conference will be held virtually on Zoom.

The conference is our annual meeting and an opportunity for young professionals dealing with different aspects of nuclear energy to present their research.

The primary objective of the meeting is to increase the recognition of young professionals and to foster cooperation amongst them.

Kindly invited!

YGN, NSS

Organizing committee:
Boštjan Zajec, Jan Kren
www.djs.si/mmg/ygnc2021
ygnc@ijs.si

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Conference Programme

Društvo jedrskih strokovnjakov Slovenije
Nuclear Society of Slovenia

8th YOUNG GENERATION NUCLEAR CONFERENCE

VIRTUAL / REACTOR CENTRE JSI, 19. May 2021

Conference opening	
8:30 – 8:45	*Conference opening*
I. Research reactors and nuclear data	
8:45 – 9:15	Calculation of gamma and neutron dose field inside the JSI TRIGA Mark II reactor hall – **Anže JAZBEC** *(invited lecture)*
9:15 – 9:30	Development of RAPID extension for TRIGA reactor 3D burnup calculations – **Anže PUNGERČIČ**
9:30 – 9:45	First-Principles Study of ZrH – **Ingrid ŠVAJGER**
9:45 – 10:00	Modelling of OPEN100 Reactor with Serpent-2 – **Anže MIHELČIČ**
10:00 - 10:15	*coffee break*
II. Fusion reactor physics	
10:15 – 10:30	Computation of Lithium Hydride Neutron and Gamma Ray Attenuation – **Andrej ŽOHAR**
10:30 – 10:45	Water activation experiment: PhD thesis topic overview – **Domen Kotnik**
III. Experimental thermal-hydraulics	
10:45 – 11:00	PIV measurements of single- and two-phase turbulent flows in a vertical pipe – **Jan KREN**
11:00 – 11:15	Measurement of turbulent flow over backward facing step with PIV – **Nejc KOSANIČ**
11:15 – 11:30	Flow boiling in horizontal annulus – visualization and bubble size distributions – **Boštjan ZAJEC**
11:30 - 12:30	*lunch break*
IV. Nuclear technology	
12:30 – 13:00	Rosatom projects worldwide and technical characteristics of VVER technology – **Alexandra OVCHARENKO, Alexander RENEV** *(invited lecture)*
V. Severe accidents	
13:00 – 13:15	Overview report on the accident at the Fukushima Daiichi NPP (summary of known facts - 10 years later) – **Kaja ZUPANČIČ**
13:15 – 13:30	Pool scrubbing simulations of SCRUPOS experiment – **Matic KUNŠEK**
13:30 – 13:45	Assessment of premixed layer formation model on experimental results – **Janez KOKALJ**
13:45 – 14:00	Computation of kinf from Monte-Carlo generated cross sections – **Jan Malec**
14:00- 14:15	*Closing remarks and coffee break*
14:15 – 15:00	**NSS Young generation network meeting**

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Notes