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Nordic Nuclear Forum for Generation IV Reactors 2014

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Nordic nuclear safety research

Abstract

The Nordic Nuclear Forum for Generation IV Reactors is a network, which has the aim to strengthen cooperation among Nordic universities and institutes regarding all aspects of generation IV nuclear reactors. Originally, the main focus was on materials research, but now areas like fuel, fuel cycles, reactor design, reactor modeling and safety are also included. The main activity has been to organize seminars, and in 2014 a two-day seminar was arranged in Lappeenranta together with Gen4Fin. The seminar had about 30 participants from seven countries; invited international experts, senior researchers, PhD-students and industry representatives. This occasion is a great opportunity for researchers to widen their personal networks and knowledge. The abstracts from the seminar are included in this report. In addition to the seminar, the activity supported a joint project between Chalmers and DTU on Monte Carlo modeling. Also the website of the forum was updated and maintained. Presentations from the seminar can be downloaded from the website. A further seminar is planned for 2016.

Key words

Generation IV, Nuclear reactors, Materials, Nuclear fuel

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Nordic Nuclear Forum for Generation IV Reactors 2014

Final Report from Nordic-Gen4

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1. Introduction

The main purpose of this NKS-R supported activity is to strengthen communication between various Nordic researchers and research groups interested in generation IV nuclear technology. The network has existed since 2009, and the coordination group consists of one member each from IFE (Norway), DTU (Denmark), VTT (Finland) and Chalmers (Sweden). The current members are Rudi Van Nieuwenhove (IFE), Bent Lauritzen (DTU), Sami Penttilä (VTT) and Mattias Thuvander (Chalmers), who is coordinator.

Generation IV (Gen-IV) nuclear fission technology aims at the development of more sustainable nuclear power, including better utilization of uranium resources, minimization of radioactive waste, as well as improved economics and safety. In the European Strategic Energy Technology Plan (SET-Plan) from 2010, maintaining European competitiveness in fission technologies and the development of Gen-IV reactors is a priority for long-term, cost-effective and low CO_2 energy production.

In 2009 a Nordic network for researchers, students and industry experts interested in Gen-IV nuclear power was established. Originally the focus was on material issues (NOMAGE4), but later the scope was widened to include all relevant areas. The main activity of the forum has been to organize seminars (typically two-day events), and to support researchers and research students making visits, in order to share knowledge as well as using unique experimental facilities. Seminars have been arranged in 2011, 2012 and 2014. It seems appropriate to have the seminars every second year, and consequently the next seminar is planned for 2016. In addition to the seminars, also smaller meetings between the participating institutions have been held, where joint activities and grant applications have been discussed. Since 2011 also a webpage is in place (nordic-gen4.org).

There is rather extensive research going on covering different areas of Gen-IV related issues, such as new fuels, fuel cycles and structural materials. The funding support from governmental bodies tends to vary over time, and for the moment the activity is strongest in Finland and Sweden, but there is both interest and competence also in Norway and Denmark. A large part of the current research is funded by the EU. Although the existing Gen-IV competence in the Nordic countries is relatively strong, there is a risk that the unstable funding situation will result in fragmentation and loss of competence that might be of great importance in the future.

Here follows a short overview of on-going Nordic Gen-IV research at universities and institutes, to illustrate the diversity of the research issues.

VTT (Finland)

VTT's Gen-IV research is focusing mainly on Super Critical Water Reactors (SCWR) and to an increasing extent also Sodium-cooled Fast Reactor (SFR) and Lead-cooled Fast Reactor (LFR) concept studies. Especially materials research has shown increased interest towards convergence of the Gen-IV material research in Europe and national level through EERA (European Energy Research Alliance) where VTT is a founding member. VTT is closely connected with Gen-IV field EU projects such as SCWR-FQT (Supercritical Water Reactor - Fuel Qualification Test), MATTER (Material Testing and Rules) and MATISSE (Materials Innovation for Nuclear Safety and Sustainability in Europe) as well as the Jules Horowitz Reactor in France. In addition, Academy of Finland has an active role in supporting Gen-IV related research of which the IDEA project is on-going (Interactive Modelling of Fuel

Cladding Degradation Mechanisms). Materials cross-cutting aspects have arisen in many projects as the main theme due to their importance in increasing efficiency. Thus, characterization of prospective candidate materials in relevant Gen-IV temperatures has been the target of many projects at VTT such as creep and creep-fatigue testing in high temperature conditions as well as oxidation and SCC susceptibility/crack growth rate tests in supercritical water.

In addition to the materials research, transition from thermal to fast reactors has been simulated lately in Finland using the French fuel cycle code COSI6 to provide transuranic inventory evolution into the future. Further, the capacity improvement for the Olkiluoto waste repository after utilising partitioning & transmutation technologies was assessed through a comparison of peak heat production by the waste in the repository in various fuel cycle scenarios.

Aalto University (Finland)

Current research at Aalto University includes simulation studies of heat transfer properties of supercritical water for SCWR, Monte Carlo burnup calculations and uncertainty analyses for sodium and lead-cooled reactors.

Lappeenranta University of Technology (Finland)

At LUT generation IV research is carried out in the field of computational fluid dynamics (CFD) for the design of gas cooled fast reactors, including the pebble bed concept, and a strong focus on nuclear safety.

Uppsala University (Sweden)

The UU research is mainly focused on safety and security. Projects that are currently running cover novel methodologies for assessing how experimental uncertainties in e.g. cross sections of relevant nuclei propagate in various nuclear systems such as fast reactors and recycling facilities, core monitoring systems for metal-cooled fast reactors, new materials for detection of ionizing radiation for core monitoring and nuclear safeguards, new methodologies and instrumentation for assessing and implementing proliferation resistance and nuclear safeguards, respectively, in recycling facilities, core diagnostics of ASTRID with the purpose to detect and diagnose anomalous operating conditions and, finally, measurement of fission yields in fast-reactor-like neutron fields and the development of codes for calculation of fission yields.

Chalmers (Sweden)

Chalmers' research is involved in several of the necessary components of a closed Gen-IV system, including separation methods for used fuel, methods for production of both oxide and novel fuels from recycled material and storage issues of the ultimate waste. In particular, Chalmers is coordinating the FP7 project ASGARD. Other projects concern fuel/cladding and fuel/cladding/coolant interaction. In collaboration with CEA Cadarache, computational methods for the neutronic modeling (spatial and energy distribution of the static neutron flux and of the neutron noise) of fast reactor systems has been developed.

In the frame of a collaboration between CEA and the Swedish research council, a multi-project entitled "Core physics, diagnostics and instrumentation for enhanced safety of the sodium cooled fast reactor ASTRID" including four PhD projects, is running.

There is also research concerning the physics, kinetics, dynamic response and neutron noise in Molten Salt Reactors (MSR).

KTH (Sweden)

The research at KTH is focused on novel nitride fuels and the lead-cooled reactor concept, including material studies. For example, new types of FeCrAl alloys with promising performance in lead at elevated temperatures have been developed together with industry. The Swedish research council will soon present an evaluation on the proposed demo and training reactor ELECTRA. Another focus area is lead-cooled small modular reactors for use in remote locations.

IFE-Halden (Norway)

IFE is developing instruments, which can be used for studying materials and fuels in Gen-IV type of reactors. Some recent achievements are the development of sensors for measuring crack growth in liquid metal, as well as the development of oxygen sensors for use in liquid metal. Further, IFE is testing various coatings for corrosion protection of materials in supercritical water and liquid metal, also in the Halden reactor.

DTU (Denmark)

Currently there is no dedicated Gen-IV research on-going at DTU. However, there is strong competence in neutronics, which is being further developed via participation in the design of ESS and ITER and there is an interest in Gen-IV research. Previously, Gen-IV projects have been carried out, for example one project on the design of heat exchangers for a suggested SCWR test loop at Halden.

2. Seminar in Lappeenranta

A two-day seminar about generation IV reactors was arranged in Lappeenranta, Finland, 4-5th September together with Gen4-Fin and LTU, with Profs. Juhani Hyvärinen and Riita Kyrki-Rajamäki as local organizers. The seminar had about 30 participants, both senior researchers and PhD-students, from seven countries (Finland, Norway, Sweden, Belgium, Russia, USA and the Czech Republic). The participants represented universities, research institutes and industry. There were several well-known invited speakers and there was also a video lecture by Prof. K. Asano, Japan. The quality of the presentations was very high. The presentations, which can be downloaded from the Nordic Gen-4 website, dealt with several different areas, such as reactor concepts, core modelling, advanced materials, diagnostics and test reactors in Europe and Japan. In addition to the presentations, a panel discussion on the topic "Future of Gen4 reactor studies and need of test and demonstration reactors" was included. Finally, an excursion to LTU nuclear safety research laboratory was organized. The number of participants at the seminar was a bit lower than anticipated. This can be related to unfortunate timing and not the best accessibility of the location, but it also reflects that the amount of research regarding generation IV in the Nordic countries is not very large for the moment. A stronger representation of fuel/fuel cycle researchers would have been desirable.

Abstracts from the presentations are listed below. Also, three presentations were given without abstracts being submitted.

1. Introduction to the Nordic-Gen4 network, Sami Penttilä, VTT, Finland.

2. The experimental investigations on liquid metal MHD heat transfer applied to fusion reactors, Valentin Sviridov, Moscow Power Engineering Institute, Russia.

3. The future of nuclear power, Juhani Hyvärinen, Lappeenranta University of Technology, Finland.

Development and testing of an oxygen sensor for use in liquid metal

Rudi Van Nieuwenhove, IFE, OECD Halden Reactor Project, Norway
Jesper Ejenstam and Peter Szakalos, KTH Royal Institute of Technology, Sweden

There is a continuing need for the development of rugged and reliable oxygen sensors for harsh industrial environments found in heat treating, metal processing and casting, glass, ceramic, automotive, aerospace, chemical and petrochemical, and food-processing industries.
Also in the nuclear industry, high temperature, radiation resistant oxygen sensors are needed in relation to the corrosion behavior of structural in-core materials. Presently, a large international research effort is going on to develop so-called Generation IV nuclear reactors which have increased safety, less radioactive waste and increased efficiency. Two of the six concepts being studied now are based on liquid metal cooling, such as liquid sodium, liquid lead or a lead-bismuth eutectic (LBE). Liquid lead and LBE are highly corrosive and corrosion control requires a careful monitoring of the dissolved oxygen.
At IFE/Halden, an electrochemical potential (ECP) sensor has previously developed for measuring the ECP of metals in water. This sensor has slightly been modified in order to function as an oxygen sensor in liquid metal. The sensor is based on a ceramic membrane made of magnesium stabilized zirconium and filled with an iron/iron oxide powder mixture. Tests at KTH have demonstrated that the sensor works reliably, providing a potential in agreement with theory. Measurement results of the first tests will be presented.

Monte Carlo method in next-generation reactor physics applications

Jaakko Leppänen, Technical Research Centre of Finland (VTT), Finland

The continuous-energy Monte Carlo method offers interesting possibilities for the modeling of next-generation reactor systems. Monte Carlo neutron transport codes are capable of modeling three-dimensional geometries with an arbitrary level of detail, using the best available knowledge on neutron interaction physics. The simulations are not limited to any specific fuel or reactor type, and the same calculation scheme can be applied to conventional LWR's, as well Gen-IV and other novel reactor concepts. The development in CPU performance and parallel computing has expanded the use Monte Carlo codes to new applications, such as spatial homogenization and coupled multi-physics simulations.

The Serpent Monte Carlo code has been developed at VTT Technical Research Centre of Finland since 2004, mainly for the purpose of reactor physics applications. The code has an active user community of more than 300 users in 114 universities and research organizations in 31 countries around the world. Typical applications range from spatial homogenization to fuel cycle simulations and research reactor modeling, many of which are related to future nuclear technologies. The code is still under development, with a strong focus on coupled calculations involving Monte Carlo neutronics and system-scale thermal hydraulics, CFD and fuel performance codes.

The challenges related to the modeling of operating nuclear reactors are introduced, along with the advantages of Monte Carlo simulation. Practical examples from Serpent users are presented, involving SFR, LFR, HTGR and MSR neutronics and coupled calculations.

Analysis and application of a non-linear consistent coupling scheme to Gen4 reactor kinetics models

Manuela Calleja, Chalmers University of Technology, Sweden

The application and analysis of an efficient coupling technique to Gen4 reactor point kinetic (PK) models is presented. To better understand the coupling between the neutronics and thermal-hydraulics it is necessary to quantify and qualify nuclear safety parameters. Moreover, even for scoping purposes. Is particularly desirable to have means to perform simple but realistic simulations. The reactor PK equations with delayed neutrons are solved in the presence of continuous temperature feedback. These equations predict the neutron density, reactivity, number of delayed neutrons and temperature profiles. All together they represent a system of coupled non-linear ordinary differential equations. Even though their applicability is limited, the PK models are useful at the level of initial studies. For this reason, it is still worth trying to develop improved schemes for the solution of these equations (mathematically characterized as stiff, non-linear, and which pose significant challenges when numerical solutions are applied). For solving theme conventional coupling schemes such as explicit approaches are used. They provide simple but inconsistent solutions by not resolving accurately the non-linear coupling terms of the equations. The work presented involves the application of a consistent coupling technique to two PK test models related to Gen4 reactors, a SFR and a MSR type.

Keywords: Multi-physics, coupled kinetics thermal-hydraulics, point models, numerical schemes, Molten Salt Reactor point kinetic model, Sodium-cooled Fast Reactor.

Remark on the calculation of the point kinetic component and dynamic response to various perturbations in the moving fuel in a MSR

Viktor Dykin and Imre Pazsit, Chalmers University of Technology, Sweden

The calculation of the point kinetic component of the neutron noise in two-group diffusion theory in Molten Salt Reactors (MSR) using different techniques is discussed. First, the point kinetic component of the noise is calculated from the full space-frequency dependent solution analytically by a projection to the static adjoint. Then, the point-kinetic solution is derived by solving the simplified point kinetic equations. Both results are thereafter analyzed and compared quantitatively. This comparison shows that the solution of the simplified point kinetic equations significantly differs from the exact one and cannot reconstruct some important features of the true solution.

The neutron noise induced by a propagating perturbation in a bare 1-D Molten Salt Reactor is calculated and analyzed using one-group diffusion theory. The neutron noise for four different noise sources including fission, absorption cross sections and fuel velocity is calculated and the results are qualitatively compared. Despite all qualitative and qualitative differences in the noise calculation procedure as well as in the noise structure, it turned out that the noise induced by absorption cross section and fission cross section follow similar behavior. It was also noticed that the inclusion of the fluctuations of the fuel velocity slightly suppresses the total neutron noise for low frequencies and enhances the latter one for high frequencies as compared to the effect of other noise sources.

The results obtained in this study bring some diagnostic significance in the interpretation and understanding of the neutron noise in MSRs.

Multiscale modelling of radiation damage production mechanisms in nuclear reactor materials

Kai Nordlund, University of Helsinki, Finland

One of the most serious problems to be solved in developing GenIV fast neutron fission and tokamak-like fusion reactor concepts is to find structural materials that can tolerate the damage from fast neutrons over prolonged operation times. The damage levels in these kind of power plants can reach 100's of dpas, and very few materials can tolerate this kind of radiation levels without unacceptably high degradation of their macroscopic mechanical properties. Moreover, MeV neutrons can, in addition to direct displacement damage from elastic recoil processes, also via nuclear reactions lead to gas production and transmutations. Hence a good understanding of the buildup of radiation damage is crucial for designing and selecting materials for future reactor concepts.

In this talk, I will overview the recent efforts to use multiscale modelling, spanning all the way from the quantum mechanics of atomic interactions to macroscopic continuum models, to understand radiation damage in materials. I will first give a brief presentation of the most widely used methods and discuss their capabilities and limitations. I will then present some recent results improving the understanding of the production of radiation damage in ferritic steels and tungsten.

Toshiba Fast reactor technology: 4S and TRU Burning Fast Reactor Cycle

Kazuhito Asano, Toshiba, Japan

The 4S (Super-Safe, Small and Simple) is a small liquid-metal fast reactor which is designed for use as a power source in remote area, intended to operate for 30 years without refueling. A pool type fast neutron has a primary electrical output of 10 MWe (30MWe). The reactor vessel is located below grade and it contains the intermediate heat exchanger (IHX), electromagnetic pumps (EMPs), internal structures, core and shielding and containment system. Heat is exchanged in a steam generator to produce steam and electricity. The 4S is based on proven and reliable technology and incorporated in innovative technology to enhance safety performance, which would be a key social infrastructure for energy supply at remote area. Plant system overview and safety characteristics are presented in this session.

On the other hand, Toshiba has been developing TRU burning fast reactor cycle using uranium-free metallic fuel. A fast reactor core using TRU fuel without uranium burns TRU with high efficiency because it does not produce any new plutonium or minor actinides. Reactor safety, cycle length and fuel fabrication are studied to review feasibility. This system is one of the most promising solutions for TRU burning considering the issues of proliferation and nuclear waste final disposal. The key issues and future developments are presented.

The Joint Programme for Nuclear Materials of the European Energy Research Alliance: an opportunity for integrated research in Europe on materials for sustainable nuclear energy

Lorenzo Malerba, The Belgian Nuclear Energy Research Centre, Belgium

The goal of sustainability is common to all low-carbon energy sources. Sustainable nuclear energy systems, or Generation 4 reactors, can be built provided that materials capable of withstanding extreme conditions like high temperature, prolonged irradiation, and chemically aggressive environments, are selected or developed and properly qualified. Some of these conditions are common to other high energy efficiency systems. Because of this pivotal importance of materials in view of sustainable nuclear energy, and innovation in the energy field in general, a joint programme for nuclear materials finds its natural place within the European Energy Research Alliance (EERA). Officially launched in November 2010, the objective of the EERA JP for Nuclear Materials is to converge towards truly integrated research activities at European level, based on the joint identification of key priority materials research topics, in support of sustainable nuclear energy systems, by building an effective collaborative framework that ensures the coordinated and optimized use of available resources and expertise all over Europe, while of course obtaining adequate funding. This presentation will discuss the level of integration reached by the JPNM by providing an overview of the structure, the objectives, the activities, the achievements, and the ambitions for the future.

Advanced materials characterization and testing for Gen IV reactor concepts

Peter Hosemann, University of California Berkeley, USA

One of the main limiting factors to realize Gen IV reactor concepts are the materials themselves and relevant data on the materials available to the engineers and designers. A lot of discussion and concepts are presented based on materials which are thought to be excellent under GENIV reactor condition but systematic quantification and detailed scientific understanding of these materials is sometimes missing due to limited access to test reactors. The candidate materials need to withstand both radiation and environments for extended periods of time making systematic studies of both necessary.
In this presentation the importance of thorough material science and new materials testing techniques allowing to accelerate materials research for GEN IV reactor applications are shown. Results of recent radiation damage and related materials studies utilizing ion beam and reactor irradiation in combination with advanced small scale materials testing are presented. Results obtained of >100dpa neutron irradiated ODS alloys using Atom Probe Tomography are discussed. Environment for especially lead bismuth eutectic cooled reactors are discussed. In addition detailed results of various steels exposed to LBE at static and flowing condition at temperatures as high as 800C are shown.

A Comparison of creep-fatigue assessment and modelling methods for GEN-IV nuclear reactor structural components

Rami Pohja, Technical Research Centre of Finland (VTT), Finland

Design codes, such as RCC-MRx and ASME III NH, for generation IV nuclear reactors use interaction diagram based method for creep-fatigue assessment. In the interaction diagram the fatigue damage is expressed as the ratio of design cycles over the allowable amount of cycles in service and the creep damage as the ratio of time in service over the design life. With this approach it is assumed that these quantities can be added linearly to represent the combined creep-fatigue damage accumulation. Failure is assumed to occur when the sum of the damage reaches a specified value, usually unity or less. The fatigue damage fraction should naturally be unity when no creep damage is present and creep damage should be unity when no fatigue damage is present. However, strict fatigue limits and safety factors used for creep rupture strengths as well as different approaches to relaxation calculation can cause a situation where creep-fatigue test data plotted according to the design rules are three orders of magnitude away from the interaction diagram unity line. Thus, utilizing the interaction diagram methods for predicting the number of creep-fatigue cycles may be inaccurate and from design point of view these methods may be overly conservative. In this paper the results of creep-fatigue tests carried out for austenitic stainless steel 316 and heat resistant ferritic-martensitic steel P91, which are included in the design codes, such as RCC-MRx, are assessed using the interaction diagram method with different levels of criteria for the creep and fatigue fractions. The test results are also compared against the predictions of a recently developed simplified creep-fatigue model which predicts the creep-fatigue damage as a function of strain range, temperature and hold period duration with little amount of fitting parameters. The Φ-model utilizes the creep rupture strength and ultimate tensile strength (UTS) of the material in question as base for the creep-fatigue prediction. Furthermore, challenge of acquiring representative creep damage fractions from the dynamic material response, i.e. cyclic softening with P91 steel, for the interaction diagram based assessment is discussed.

Project ALLEGRO: A Helium-Cooled Fast Reactor Demonstrator

Ladislav Belovsky, UJV Rez, Czech Republic

The goal of the ALLEGRO Project is to design, build and operate the first Gas cooled Fast Reactor (GFR) Demonstrator. The GFR in Europe represents together with the Lead cooled Fast Reactor (LFR) a longer timescale alternative to the Sodium Fast Reactor (SFR), all of them belonging to the 4th generation of nuclear reactors. The original concept of ALLEGRO was designed in France by CEA in 2002-2009 with the aim to develop a high temperature (850 ºC) demonstration unit suitable for testing the new U-Pu carbide fuel in ceramic SiC/SiCf claddings in prototypic helium coolant conditions as well as the GFR-related technology. Because ceramic fuel is still under development, MOX fuel was chosen for the first ALLEGRO core producing helium temperatures of 530 ºC. The ALLEGRO design studies have been shared in the GCFR 6th FP since 2005 and in the GoFastR 7th FP since 2010. As France gave priority in 2009 to the development of the SFR demonstrator ASTRID, CEA proposed the continuation of the ALLEGRO project to four institutes from Central Europe: MTA-EK (Hungary), ÚJV Řež, a. s. (Czech Republic), VUJE a.s. (Slovak Republic) and NCBJ (Poland).

The presentation gives insight into the continuation of the ALLEGRO project since 2010. MTA-EK, ÚJV Řež, a. s. and VUJE a.s. signed Memorandum of Understanding in May 2010 that defines the basic documents to be prepared during the so called Preparatory phase of the ALLEGRO project before 2020 that would facilitate the decision whether to start the detailed design and build the demonstrator. Association "V4G4 Centre of Excellence" of the four mentioned CE entities was established in 2013 with the aim to provide a legal structure for fulfilling the Preparatory phase of the project. Three CE countries wish to host the ALLEGRO reactor and will prepare the basic project documents by using their own financial resources, in combination with expected governmental support in their countries as well as with a potential international support from the EU Framework Programs and/or EURATOM Generation-IV funds. Regulators of the three countries and the French regulator will be invited to participate in the Preparatory phase of the project by reviewing the Licensing roadmap, content of the Preliminary safety analysis and the Criteria of siting.

CFD modelling supporting the design and development of gas cooled fast reactors

Heikki Suikkanen, Lappeenranta University of Technology, Finland

Efficient heat transfer between the fuel and the coolant requires additional measures in gas-cooled reactors. Helium is the gas typically considered as the coolant medium for modern gas cooled reactors due to its rather good properties compared to other coolant gas candidates. Helium needs to be pressurised to 7-10 MPa to improve heat transfer in the reactor core. One way to further improve heat transfer is to cover the heat transfer surfaces with roughness structures that induce turbulence and thus enhance heat transfer with some expense to the pressure losses. Optimisation of the roughness structures is needed, as the pressure losses through the reactor should be minimised to reduce pumping costs and to ensure natural circulation in abnormal situations.

Experimental and numerical work has been conducted within the European project THINS (Thermal-hydraulics of innovative nuclear systems) to investigate the pressure losses and turbulent heat transfer in a single heated fuel rod geometry fixed inside a hexagonal flow channel. This geometry of the test section of the L-STAR experimental facility represents the sub-channel of a single fuel rod of the ALLEGRO gas cooled fast reactor. The presented work covers the CFD simulations of the L-STAR experiments that were performed with the ANSYS Fluent CFD code.

A rod with a smooth surface and one with square cross-section ribs attached on the surface were investigated. A three dimensional steady-state calculation model was constructed of the L-STAR test section and various RANS (Reynolds-averaged Navier-Stokes) turbulence models available in the CFD solver were tested. The realisable $k-\epsilon$ model was found out to be most applicable for the problem and gave reasonable results in the case of the smooth rod when compared to experimental data. However, in the case of the roughened rod the model under estimated the heat transfer severely. Based on the results obtained, further studies with, for example, non-linear eddy viscosity or Reynolds stress turbulence models are suggested before utilising CFD for making credible conclusions regarding the heat transfer performance of roughened rod configurations.

Pebble bed reactor modelling at LUT

Ville Rintala, Lappeenranta University of Technology, Finland

Pebble bed reactors are graphite moderated gas cooled reactors where fuel is inside of graphite spheres with typical diameter of 6 cm. Fuel is in the form of small particles and thousands of these fuel particles are needed for each pebble. A few hundred thousand pebbles are needed for power reactor and pebbles are randomly arranged in reactor core. On the other hand fuel particles are placed randomly to fuel zone of the pebbles and as such there is random structure on two levels which has been named as double heterogeneity problem.

In LUT the positions of fuel pebbles are calculated with discrete element method (DEM). Resulting location information is transferred to reactor physics and used to calculate porosity needed in thermal hydraulics. Reactor physics calculations are made with Serpent continuous energy Monte Carlo code and described double heterogeneity problem is modelled explicitly. For thermal hydraulics general purpose CFD (Computational Fluid Dynamics) code Ansys Fluent is used with porous media approach. For coupled analysis some preliminary coupling has been already made with Serpent and Fluent and more accurate work is ongoing.

Flow measurement using Wire-Mesh Sensor technique

Arto Ylönen, Lappeenranta University of Technology, Finland

Wire-Mesh Sensor (WMS) is a well-established method for studying single- and two-phase flows. The measurement technique was developed by Prasser et al. (1998) and has been applied to many different channels and geometries to investigate flow behavior and geometry-related flow phenomena. The working principle of the sensor is based on the measurement of electrical properties of the fluid in-between the two wire layers of the sensor. Conductivity Wire-Mesh Sensors are the most widely used WMSs, but quite recently developed capacitance WMSs have gained popularity as they can be applied to study behavior of non-conductive and high viscous fluids such as two-phase flow of silicone oil and air.
Despite of being an intrusive measurement technique, WMS enables the detailed measurement of flow with high spatial and temporal resolutions. Typically spatial resolution (wire pitch) is above 1.5 mm and temporal resolution can be up to 10 000 frames/s (16 transmitter wires). These numbers make it superior compared to point-wise measurement techniques (local conductance and optical probes).
Some examples of different types of WMSs are presented and more detailed results are shown from two selected experimental facilities, SUBFLOW (PSI, Switzerland) and HIPE (LUT, Finland).

Particle image velocimetry (PIV) measurements at LUT nuclear safety research

Joonas Telkkä, Lappeenranta University of Technology, Finland

Particle Image Velocimetry (PIV) is a non-intrusive measurement technique to study flow fields. With PIV one can obtain two-dimensional or three-dimensional velocity vector fields from a measurement plane or a space. PIV can be used to measure both one-phase and two-phase flows and for example flames. In order to make a PIV measurement one needs tracer particles which follow the flow as faithfully as possible, a light source to illuminate the seeding particles and a camera system to record the motion of the particles. This motion is used to calculate the speed and direction – it is the velocity field – of the flow being studied.

At LUT nuclear safety research PIV has been used for two-phase flow studies in the PPOOLEX test facility, which models the suppression pool of a boiling water reactor. Main goal of these studies is to produce validation data for CFD (Computational Fluid Dynamics) calculations. The focus has been on measuring the flow field in the vicinity of the blowdown pipe mouth inside the suppression pool. There are two main schemes that have been measured: stationary condensation, where the steam is inside the blowdown pipe and more chaotic situation, where bursts of steam bubbles in various sizes condense into the suppression pool. The stationary situation is fairly easy to measure, whereas the chaotic scheme is quite challenging. This is due to reflections from the steam bubbles, which makes phase separation mandatory. Also optical distortions are present. In addition to that, there are high velocity gradients between the two phases.

PIV system used at LUT nuclear safety research is presented. Measurements conducted with the system are shortly introduced.

3. Other activities

3.1. Research project on "Stationary Fluctuations"

As a part of the network project, a master student (Eirik E. Pettersen) enrolled at DTU carried out an internship project at Chalmers under the supervision of Prof. C. Demazière and PhD-student K. Jareteg. The project was very successful and resulted in an internal report and a paper with the title "Development of a Monte-Carlo Based Method for Calculating the Effect of Stationary Fluctuations" has been submitted to the conference "ANS MC2015 - Joint International Conference on Mathematics and Computation (M&C), Supercomputing in Nuclear Applications (SNA) and the Monte Carlo (MC) Method - Nashville, TN, April 19-23, 2015, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2015). The authors of the paper are E.E. Pettersen, C. Demazière and K. Jareteg (Chalmers) together with T. Schönfeldt, E. Nonbøl and B. Lauritzen (DTU).

3.2. Meetings

Prior to the seminar in Lappeenranta a meeting was held at Arlanda Airport on the 12th of June. At the meeting some questions concerning the up-coming seminar, in particular the choice of invited speakers, were discussed. Also, the participants (from IFE, DTU, VTT, Chalmers, KTH and UU) briefed each other about on-going generation IV research at their respective institutes/universities.

3.3. Website

The website www.nordic-gen4.org has been maintained during the year. The presentations from the seminar can be downloaded from the website.

3.4. Logo

Following a discussion in the coordination group, a new logo was designed, see figure 1.

Figure 1. The new logo for the Nordic-Gen4 network.

4. Conclusions

The main task of this activity was to organize a seminar bringing people from different areas of nuclear generation IV research in the Nordic countries together. The seminar in Lappeenranta was clearly successful in terms of the program and the quality of the presentations. However, the number of participants was lower than expected. The plan is to organize a seminar again in 2016, and then efforts are needed to reach a larger audience. The other activities undertaken have served to keep and improve the connections among the participating partners.

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Abstract

The Nordic Nuclear Forum for Generation IV Reactors is a network, which has the aim to strengthen cooperation among Nordic universities and institutes regarding all aspects of generation IV nuclear reactors. Originally, the main focus was on materials research, but now areas like fuel, fuel cycles, reactor design, reactor modeling and safety are also included. The main activity has been to organize seminars, and in 2014 a two-day seminar was arranged in Lappeenranta together with Gen4Fin. The seminar had about 30 participants from seven countries; invited international experts, senior researchers, PhD-students and industry representatives. This occasion is a great opportunity for researchers to widen their personal networks and knowledge. The abstracts from the seminar are included in this report. In addition to the seminar, the activity supported a joint project between Chalmers and DTU on Monte Carlo modeling. Also the website of the forum was updated and maintained. Presentations from the seminar can be downloaded from the website. A further seminar is planned for 2016.

Key words: Generation IV, Nuclear reactors, Materials, Nuclear fuel

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