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The Finnish Sustainable Energy (SusEn) project on New Type Nuclear Reactors (NETNUC)

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1 ABSTRACT

A considerable international effort is being devoted to the development of new reactor concepts, known as Gen IV reactors, which differ essentially from the existing light water reactors. The new reactor types raise new engineering challenges with new operating regimes and safety features. In order to benefit of these new reactor technologies, it is necessary for Finland to join international projects and develop domestic expertise on the technologies involved. The NETNUC project is a multidisciplinary consortium to carry out basic research to generate scientific knowledge needed for Gen IV reactors and to educate a new generation of research scientists in the field. The consortium partners are LUT, TKK and VTT.

Main research areas in LUT are the supercritical water reactor (SCWR) and gas-cooled reactors. The SCWR research is linked to the existing experimental work on light water reactor thermal hydraulics. The gas-cooled reactor research is a new field in LUT and consists mainly of numerical modelling activities. Research in TKK includes thermal-hydraulics of SCWR and Monte Carlo calculations on reactors utilizing thorium fuel. The research focus at VTT is on fast reactors, SCWR materials testing, and the use of nuclear heat in integrated industry.

2 INTRODUCTION

Increased international attention has recently been devoted to reactor concepts that differ essentially from the existing light water reactors (LWR). Basic processes of these new concepts, known as Gen IV reactors, are fundamentally different from those used today. Some new features may create new type safety challenges. They also aim to push nuclear reactor technology to completely new regimes of performance parameters, thus raising engineering challenges.

In order to allow Finland to benefit from these new technologies and influence their development, it is necessary to join relevant international projects, develop domestic expertise on critical technologies involved, and participate in ongoing international efforts to develop safety requirements for them. The New Type Nuclear Reactors (NETNUC) project, as a part of Finnish Sustainable Energy (SusEn) research programme, is a multidisciplinary consortium to carry out basic research to generate scientific knowledge needed for Gen IV reactors and to educate a new generation of research scientists in the field. The research work is carried out in tight co-operation between the project partners LUT, TKK and VTT.

Main research areas in LUT are the supercritical water reactor (SCWR) and gas-cooled reactors, the latter being a new field in LUT. Linking of the strong thermal hydraulic background of LUT to the safety research of reactors operating in supercritical pressure conditions has been studied. Computational fluid dynamics (CFD) calculations have been carried out to simulate the boiling water reactor pool tests of LUT, which are relevant also for SCWR. The research of gas-cooled reactors in LUT has focused on the numerical modelling of coolant flow and heat transfer in the reactor core of a pebble-bed type high-temperature gas-

cooled reactor. There has been co-operation with a research group in LUT that has experience in the modelling of combustion processes and gas-phase.

Main research areas for TKK are the thermal-hydraulics of SCWR and Monte Carlo calculations on reactors utilizing thorium fuel. In addition to this the different aspects of the fuel cycle have been analyzed. SCWR thermal hydraulics has been investigated by simulations with APROS process simulation software. Also co-operation with Kurchatov Institute, Russia has been initiated. Thorium breeding in nuclear reactors has been investigated with MCNP simulations. Several auxiliary programs for thorium fuel performance calculations with FEMAXI-6 have been developed.

Main research areas in VTT are the fast reactor and the SCWR concepts. In 2008, the first calculation system capable of analysing fast reactor concepts was taken into use at VTT. The validation of this system was started as well. The work included development of calculation methods by creating a new burn-up calculation method based on calculating matrix exponentials. In the field of supercritical water cooled reactors, the participation in the EC HPLWR2 project was continued with development of thermal hydraulics models and applying them to the safety studies of the high performance light water reactor, Kurki and Seppälä (2009).

Understanding of corrosion phenomena of candidate materials under SCWR conditions necessitates a reliable experimental testing of materials and therefore also a development of monitoring techniques for the relevant conditions, Penttilä et al. (2009a). The long term objective of this project is to perform a state-of-the-art study that would serve as a guide for the selection of in-core materials for the SCWR. One of the key performance indicators for the material selection is the corrosion and oxidation behaviour of materials in SCWR conditions. Therefore, in situ studies of the oxide films forming on the internal component candidate materials in contact with supercritical water are needed. In 2008, first steps towards employing a pneumatic servo-controlled bellows system for the oxidation studies under SCWR conditions were performed. The pneumatic loading technology provides important potential benefits and has already been successfully applied in different test environments. Calibrations and installations of the double-bellows system into the supercritical autoclave have been performed. Usability of the double-bellow system was verified up to 650°C and 25 MPa.

In the future, it is likely, that process heat and steam from new type high-temperature nuclear power plants is used in the integrated industry, such as bio-refineries. It is important that also simulation tools support processes with multi-phase chemistry. The feasibility of combining rigorous multi-phase chemistry with the dynamic process simulator, APROS, was evaluated at VTT. Three different case processes were studied: fibre suspension and bleaching chemistry in pulp and paper mills, and boiler water chemistry. The multi-phase chemistry and process simulation were successfully combined.

3 GAS-COOLED REACTOR RESEARCH AT LUT

The very high-temperature reactor based on gas-cooled reactor technology is perhaps the first Gen IV concept going to be demonstrated. There are ongoing reactor projects, such as PBMR in South Africa, Koster et al. (2003), and HTR-PM in China, Zhang et al. (2009), which will, if successful, prove the feasibility of the concept. The reactor concept has interesting features like passive safety due to its heat transfer characteristics, and the possibility to generate high-temperature heat for industrial process applications. Characteristic for the reactor type is the use of helium coolant and coated particle fuel dispersed in graphite. There are two main core designs, the prismatic core with fuel particles inside block type fuel elements and the pebble-bed core with fuel particles inside graphite spheres. The power plant cycle can be direct or indirect, latter allowing the use of different fluid in the secondary circuit.

In LUT, the research activities have been focusing on modelling a pebble-bed core. The efforts include studies on packing and flow of fuel pebbles, reactor physics, coolant flow, and heat transfer. The goal is to establish reliable modelling methods, which can be used for core analysis in all possible situations. As a by-product, the national knowledge level on this specific reactor design increases and the methods developed could contribute to the research of other reactor types or even non-nuclear applications. In addition to modelling, there are preliminary plans to launch some experimental research activities related to gas flows and heat transfer. There is a versatile transparent experimental facility in LUT, which is primarily considered for LWR research, but could be modified for use in gas-cooled reactor research.

A typical pebble-bed reactor core design consists of a random pack of spherical fuel elements in a cylindrical annulus between graphite reflectors. The number of fuel pebbles in the core can be several hundreds of thousands. A pebble is fed to the core from the top, it travels trough the core, and is removed via defuel chutes. After its removal from the core, the pebble is analyzed and it can either be returned to the core or removed from use. The locations and residence times in different parts of the core vary between the fuel pebbles, so each one has a unique burn-up history. The packing structure affects coolant flow and heat transfer. This effect can be observed, for example, near the walls where the packing density is smaller than the core average and leads to coolant flow channelling. The pebbles add to the stresses of structural materials and are themselves affected by the mass of each other. The friction between touching pebbles and between pebbles and reflector walls causes some graphite dust to be formed, which may have some effect on heat transfer and some other properties. As the core meltdown is an excluded accident in this reactor concept, the source term for radioactive release from the graphite dust containing activity is what needs to be considered. For these reasons, the research of packing and flow of fuel pebbles is essential.

The research of fuel pebble flow and packing belongs to the research field of granular materials. There is previous experience in the research of granular materials in LUT, see for example, Jalali et al. (2003 and 2006). In the current pebble-bed research, discrete element method (DEM) is used for detailed modelling of pebble flow and packing. The actual tools mostly include in-house developed codes and open source software. In a DEM analysis the forces acting on individual pebbles are resolved. The data that can be extracted from the analysis includes, for example, locations and velocities of each pebble at a certain time. The data can then be used in other parts of the modelling; for example, the pebble position data can be used to generate an averaged distribution for the packing density to be used in full core coolant flow calculations. The friction forces between pebbles can be used to develop models for dust production and thus, the source term for radioactive release. Figure 1 shows an example of a DEM simulation done in LUT.

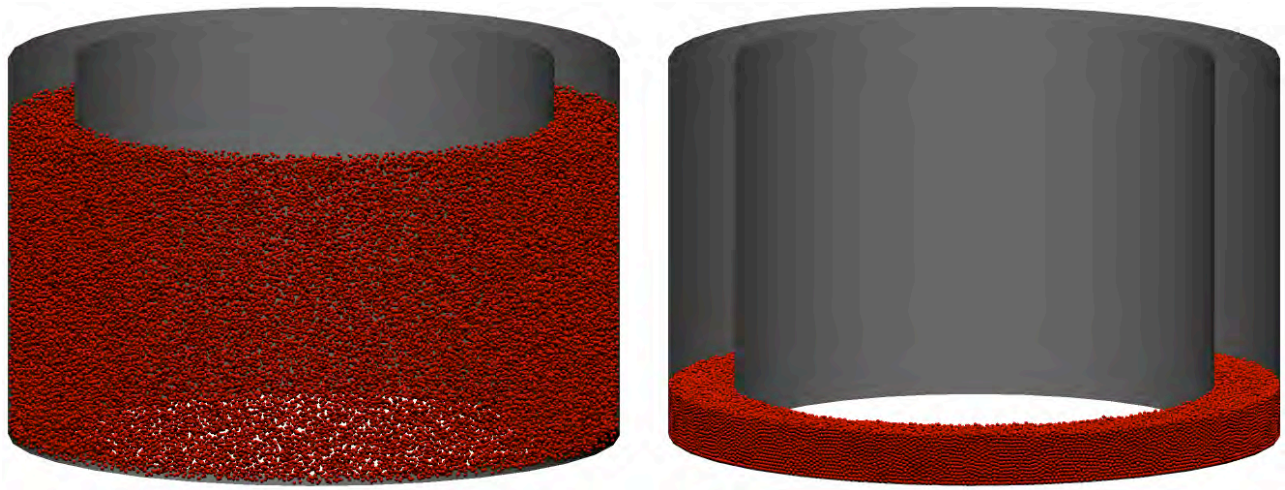

Figure 1. An initial configuration of approximately 100 000 spherical particles inside an annular cylinder (left) is let to collapse due to gravity in a DEM simulation. The average packing density of the spheres in the initial configuration is 0.10 and after 0.56 seconds it increases to approximately 0.63.

Reactor physics modelling of a pebble-bed reactor is a challenging task, not only because of the huge number of randomly distributed coated fuel particles inside the fuel pebbles, but because the fuel pebbles are also packed randomly in the core. Because of the random nature of the configuration, a stochastic approach using Monte Carlo method has been selected for reactor physics studies. The method allows the use of as detailed geometry as necessary. The drawback is the huge requirement of computing power. The Monte Carlo reactor physics code Serpent, developed at VTT Technical Research Centre of Finland, Leppänen (2007a), is used for reactor physics studies at LUT. Serpent has several ways to handle the double heterogeneous fuel of pebble-bed reactors. One way is to use a regular lattice of fuel pebbles in the core and a regular lattice of coated particles inside the pebbles. Some homogenization could be used if reduced computing time is required. It is also possible to take the random locations of fuel particles into account by using a randomly dispersed particle fuel model in Serpent, which utilizes the possibilities available in the Monte Carlo method, Leppänen (2007b). Yet another alternative is to define the fuel particles and even the fuel pebbles explicitly in random configurations. This is demonstrated in Figure 2. The different geometry

options in Serpent are currently being tested and the preliminary results show good performance in pebble-bed calculations.

Coolant gas travels trough the pack of fuel pebbles carrying away the thermal energy generated in the fuel. Resolving the flow field and temperature distribution in the core is important, especially for reliable safety analyses. There are many details, such as leakage flows, that can be included in thermal hydraulic calculations and for that reason CFD is used as a method. CFD can be used for very detailed analyses of individual components and regions of just a few fuel pebbles, or with some simplifications for whole core analyses. Work has already been done in modelling steady state thermal hydraulics of a pebble-bed core using porous media models to represent the fuel pebble packing, Suikkanen (2008). Commercial CFD software Fluent with some user coding has been used as the tool in the calculations so far, but there is a growing interest in moving towards open source CFD codes with more freedom to include own models.

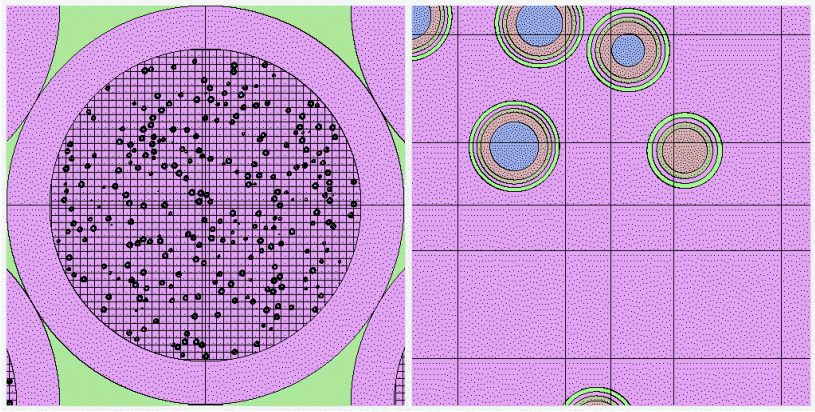

Figure 2. Randomly dispersed coated fuel particles inside a fuel pebble (left) and a magnification of individual fuel particles (pictures taken with the plotting tool in Serpent).

The numerical methods that are used in the current research work represent the state-of-the-art in modelling, but they also require lots of computing power and are dependent on parallel computing. Coupling of the separate codes can result in very detailed information and can later be used for the validation of faster running simplified codes. In addition to the obvious coupling of reactor physics and thermal hydraulics, the fuel pebble location information from a DEM code can be used for generating a profile for packing fraction to be used in porous media based CFD analysis or the exact positions can be used in cases when a detailed random geometry of a region with just a few pebbles is needed. Similarly, the positions can be used to generate a detailed geometry for Monte Carlo reactor physics calculations. It is also possible to use the data from a DEM simulation in structural analyses of the reactor vessel.

4 THORIUM FUEL CYCLE ISSUES REVISITED

The price of uranium is expected to fluctuate both due to global market situation and to supply-demand discrepancies. As several new nuclear power plant units are being started and their operating life is getting close to 60 years and even beyond, the ^{235}U-resources may become critical in the long run. Therefore, TKK has studied the viability of thorium assuming the current nuclear electricity infrastructure and LWR power plants.

Thorium is three times more abundant in the earth's crust compared to uranium. It appears as the fertile ^{232}Th isotope, which can be converted into fissile ^{233}U by exposing it to neutron irradiation in a nuclear reactor. Th/^{233}U fuels are superior to the traditional ^{238}U/^{235}U fuels in many ways. For example, the heat conductivity and chemical stability are more favourable, the thermal conversion ratios are high (even thermal breeding has been demonstrated) and the discharged fuel contains less transuranic elements, which eases the disposal of the fuel. It has also been found out by Kang and von Hippel (2001) that proliferation risks are smaller in thorium-based fuels because of the inevitable presence of denaturing ^{232}U. Its decay chain contains strong gamma emitters, which makes processing of the discharged fuel quite complex.

Unfortunately, the presence of ^{232}U disturbs the non-proliferative uses of the fuel. The strong gamma radiation together with poor solubility originating from the chemical stability make reprocessing of the thorium fuels more complicated than that of uranium fuels. This, of course, affects also the economical

feasibility of closed thorium cycles. On the other hand, feasible once-through thorium cycles require high burn-up compared to the presently applied license limits.

At the moment, a quite promising once-through thorium fuel concept called Radkowsky fuel bundle, Galperin et al. (1997), is being studied using coupled Monte Carlo and burn-up calculation code Serpent, Leppänen (2007a). The aim of this study is to find out, whether it would be advantageous to substitute part of the fuel bundles of an ordinary pressure water reactor with Radkowsky bundles or not.

Thorium fuel performance in thermal reactors is superior to uranium fuel in many aspects. FEMAXI-6 fuel performance code, Motoe et al. (2006), has been updated in our studies with thorium material properties from open literature. Most correlations do not take into account burn-up dependent phenomena, which invalidates the analysis for higher burn-up. However, the observed correlation between simulated and experimental (OECD/NEA Halden project, VTT) fuel behaviour is promising in the low burn-up region.

5 MATERIALS PERFORMANCE AT SUPERCRITICAL WATER (SCW) CONDITIONS

The SCWR concept is very attractive due to the fact that it offers a high efficiency and the basic concept is close to the present LWRs. Added to this, many components can be adopted directly from supercritical water fossil power plants. Although new concepts will offer attractive features, at the same time they also bring new and demanding challenges e.g. for the complicated core reactor physics and hydraulics design as well as for the materials technology due to increased operating temperatures and irradiation doses as well as more aggressive coolants and/or longer life time expectations than in Gen II and Gen III plants. The Gen IV reactors are built to last for ~60 years so the viability of the SCWRs depend on the capability of the structural materials of the reactor to withstand the harsh environment for the required time. The coolant outlet temperature is about 500°C and the fuel cladding peak temperature can be as high as 620 - 630°C, setting the fuel cladding to be one of the most difficult components to be handled. Zr-alloys used in present day LWRs oxidise too quickly at the SCWR temperatures. The choices will be mainly a modified stainless steel or novel oxide dispersion strengthened (ODS) alloy. The other core components are operating at up to 500°C, which is in the upper limit of conventional austenitic stainless steels.

The goal of this project is to model the general corrosion processes of the SCWR core materials. The oxidation behaviour of candidate materials will be studied using both in-situ and ex-situ methods and applying backward calculation to model materials oxide structure and composition. Additional studies are stress corrosion cracking (SCC) susceptibility and creep behaviour tests of the selected materials.

Several publications available include results from general corrosion and SCC studies of candidate materials for SCW, Lee et al. (2009), Hwang et al. (2008), Peng et al. (2007) and Saito et al. (2006). Tests have been conducted in a wide range of conditions. However, temperatures used in the experiments, for both general corrosion and SCC, have been systematically lower than the estimated peak temperature of the cladding, i.e. ~650°C. Candidate materials for SCWR are ferritic-martensitic steels, including ODS ones, austenitic stainless steels and Ni-based alloys, Penttilä et al., (2008 and 2009b). Generally, the tested materials have shown similar trend, i.e. the weight gain increases with increasing temperature. The results show that the most corrosion resistant alloys under SCW conditions are high chromium (> 18%Cr) and high nickel (> 15%Ni) alloys, which include PM2000, some of the stainless steels, e.g. 800H, and Ni-based alloy 625.

Many material properties are needed as input data for Gen IV concept design work and generation of reliable material data is needed. At VTT, the contact electric impedance (CEI) and the contact electric resistance (CER) techniques have been successfully used for in situ characterization of the electrical and transport properties of the corrosion layers both in LWR and SCWR conditions that usually is electrically very poorly conductive environment. Based on previous CEI measurements, the oxide on pure Ni has behaved as an insulator, whereas that on pure Cr as a p-type semiconductor in the SCW conditions at 500°C, 30 MPa and DO_2 = 0 – 2.3 ppm, Betova et al. (2006). Conversely, the resistances of the films on Fe and 316NG stay low, indicating a conducting or n-type semiconducting behaviour. This means that the prevailing oxide is magnetite type. This result is in line with the conclusion from recent CER/CEI measurements done with the austenitic alloys 316NG, 1.4970 and 347H. Again, results have indicated that ionic transport processes can not be distinguished from commercial austenitic steels due to high electronic conductivity of the films under SCW conditions (T = 500-650°C, p = 25 MPa, DO_2 = 150 ppb – 1.0 ppm).

The pneumatic loading technology has already been successfully applied for materials testing in different test environments, Moilanen (2004). The pneumatic servo-controlled material testing system has earlier been used to perform fracture toughness, corrosion fatigue and electrochemical measurements in gas, high-temperature aqueous and irradiation environments. The main advantages of pneumatic testing system are sensitivity and possibility to separate material testing and control system by long pressure tubes. Also, the system can be made so small that testing system can be fitted to a very small volume. A new double-bellows pneumatic loading unit has been installed to supercritical autoclave at VTT and usability of the double-bellows system has been verified up to 650°C and 25 MPa. It has been demonstrated that double-bellows system is technically feasible to carry out well defined and controlled material testing at SCW conditions. This makes possible to study both mechanical properties and oxidation resistance of the candidate materials. The preliminary test results indicated that the double-bellows system works reliably and that well defined and controlled load can be generated. However, it was decided to use Ni base alloys for pressure tubes due to low oxidation resistance of the austenitic alloy tube material (316L) in SCW. As a next step in the end of 2009, crack growth tests and CER measurements will be done under SCW conditions (650°C, 25 MPa).

In order to understand the behaviour of the metal/oxide/film coolant system on a theoretical level, physico-chemical modelling is needed. The thickness and the in-depth composition of the oxide films formed on candidate materials at SCW conditions will be estimated using ex-situ methods, e.g. glow discharge optical emission spectroscopy. This information on the oxide film thickness and composition can be used as boundary conditions for calculations of transport properties in the different layers of an oxide film. The calculations procedure is based on general physical and (electro)chemical approaches and comprises of a quantitative treatment of the growth of the inner, or barrier oxide film layer via solid state transport of matter and charge mediated by the generation, diffusion-migration and consumption of point defects (interstitial ions and vacancies) and possibly also a diffusion formalism for the growth of the crystallites of the outer (porous) layers.

6 NUCLEAR HEAT RESEARCH AT VTT

There is increasing demand for implementing the rigorous multi-phase chemistry as a part of full-scale process simulation. During the last decade, several large-scale process models have been developed using APROS process simulator without implemented multi-phase chemistry as a part of the modelled process. There is an apparent need for implementing precise chemistry in such processes where nuclear heat is exploited in chemical and forest industry. The usability of these process models is limited if no chemistry is implemented. The aim of this study was to evaluate the feasibility of using the multi-phase chemistry as part of large-scale process simulation.

A typical multi-phase system, in this case the pulp suspension, shown in the Figure 3, can contain following phases and compounds:

- Gas phase: $O_2(g)$, $N_2(g)$, $H_20(g)$ and $CO_2(g)$
- Aqueous phase: $H_2O(aq)$, $H^+(aq)$, $OH^-(aq)$, $Ca^{+2}(aq)$, $CaCO_3(aq)$, $CH_3COO^-(aq)$, $CH_3COO^-(aq)$, $CH_3COOH(aq)$, $Cl^-(aq)$, $CO_2(aq)$, $CO_3^{-2}(aq)$, $HCO_3^-(aq)$, $HSO_3^-(aq)$, $HSO_4^-(aq)$, $Na^+(aq)$, $SO_2(aq)$, $SO_3^{-2}(aq)$, $SO_4^{-2}(aq)$, $AcidFree_{7.2}^-(aq)$ and $AcidFree_{7.2}H(aq)$
- Fibre phase: $H_2O(f)$, $H^+(f)$, $OH^-(f)$, $Ca^{+2}(f)$, $CaCO_3(f)$, $CH_3COO^-(f)$, $CH_3COOH(f)$, $Cl^-(f)$, $CO_2(f)$, $CO_3^{-2}(f)$, $HCO_3^-(f)$, $HSO_3^-(f)$, $HSO_4^-(f)$, $Na^+(f)$, $SO_2(f)$, $SO_3^{-2}(f)$, $SO_4^{-2}(f)$, $AcidFree_{7.2}^-(f)$, $AcidFree_{7.2}H(f)$, $Acid_{2.1}^-(f)$, $Acid_{2.1}H(f)$, $Acid_{4.8}^-(f)$ and $Acid_{4.8}H(f)$
- Solid phase: $CaCO_3$

The implementation of multi-phase into the large-scale process simulation was evaluated using three different case processes: pulp suspension in the short circulation of paper machine, boiler water in boilers, and pulp suspension in the pulp bleaching line. The short circulation of paper machine is shown in Figure 3.

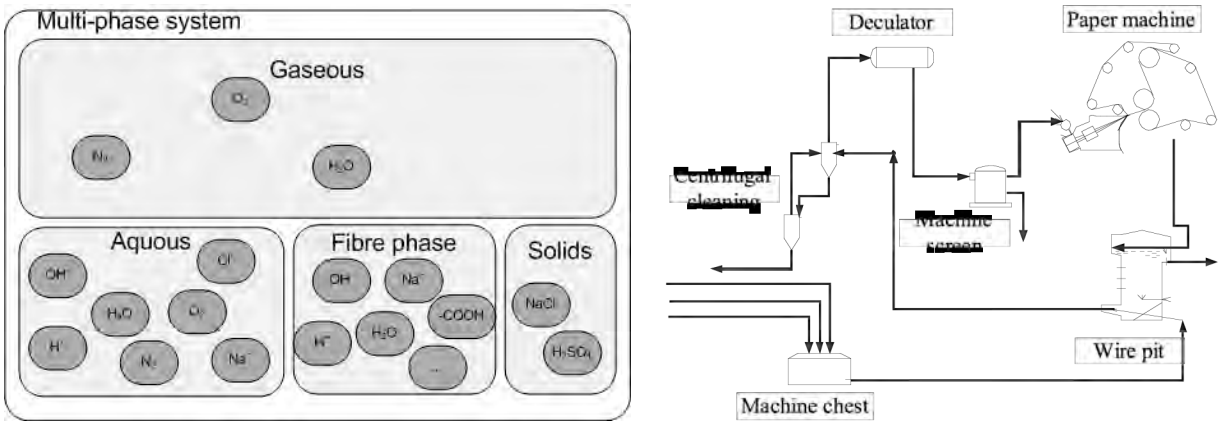

Figure 3. A typical multi-phase system with four separate phases and dozens of different compounds (left), and an example case for implementing multi-phase chemistry in the large-scale process simulation, such as the short circulation of a paper machine, Kangas (2009).

Three different approaches to calculate thermodynamic equilibrium of a multi-phase system were evaluated. Equilibrium constants, Gibbs' free energy minimization technique, and neural networks were studied. The aim was to find a suitable calculation method to be used as part of large-scale process simulation. A dynamic process simulator, APROS, was used for process modelling. Mass and energy balances, all the unit operations and streams were defined inside the process simulator as well as the needed automation. Process model included tens of calculation volumes, where pressure and enthalpies as well as mass fractions were solved. Typical calculation step for this kind of process is 0.2 seconds. During every calculation step there can be one or several iterations, when thermodynamic equilibrium is calculated. This leads to hundreds or thousands of equilibrium calculations during every second. And for the automation tuning and training purposes, the process simulator should run at least real time. This set high demands for the efficiency of calculations.

It was possible to use multi-phase chemistry as part of the large-scale process simulation with some restrictions. Process simulator, APROS, did not support the concept of multi-phase chemistry. Multiple phases are not included in the standard architecture of the program. Thus, the number of flowing compounds was too low for modelling a rigorous multi-phase system. Either simplification of chemical system or development of simulation software was needed. The interfaces between the process simulator and multi-phase engines need to be improved. Currently there are only different legacy standards, which are not compatible.

Gibbs' energy minimization seems to provide the most universal approach for calculating the thermodynamic equilibrium as part of a large-scale dynamic process simulator. Equilibrium constants can be used if temperature dependence is covered. Neural networks can be used in some special circumstances. They are optimal for soft-sensor type applications. Calculation speed is still the limiting factor. By using current desktop workstation it is not possible to perform large-scale real-time simulations. Optimizing "research" code can offer remarkable speed additions, but parallel computing could provide bigger possibilities in the near future.

Next step is to develop a BioNuclear-concept where nuclear heat is exploited in a bio-refinery in order to produce bio-based chemicals and fuels from biomass. This is the main focus of the study during the year 2009.

7 CONCLUSION

The NETNUC project as an active part of SusEn research programme consists of the researchers linked together by the national Finnish research network GEN4FIN. The results of NETNUC are disseminated through GEN4FIN to the Finnish industry. VTT has also been active in the preparation of the strategic research agenda of the Sustainable Nuclear Energy Technology Platform. Also Nordic co-operation on Gen IV field has been started – the NOMAGE 4 network funded by NKS is participated by industrial companies and other partners and co-ordinated by Studsvik Company from Sweden. The work will be linked with NETNUC.

SCWR work has continued in VTT in tight connection with the Euratom research project HPLWR2. In LUT the SCWR research is linked to boiling water reactor containment wet well condensation studies. Candidate materials for SCWR conditions have been tested with novel experimental methods by VTT. Understanding of the oxide layer properties of the candidate materials on a theoretical level is essential for predicting their long-term behaviour. Thus theory, modelling, and experimental work together can provide the proper approach to material issues in SCW environment.

The gas-cooled reactor research in LUT has started with modelling activities related to pebble-bed reactors and with participation in courses organized in EU within the context of Raphael project. Participation to forthcoming European research projects related to gas-cooled reactors is planned.

Thorium fuel cycle possibilities in current LWR reactors have been studied in TKK. For example, calculations on Radkowsky fuel bundle have been made using Serpent Monte Carlo code. The research on thorium fuel cycle issues will continue in TKK.

As Gen IV (and even Gen III) reactors provide new possibilities of process integration, where nuclear heat and steam is exploited by the nearby process industry, it is needed to improve the calculation capabilities of process simulators. Research work has been done to implement rigorous multi-phase chemistry into the large-scale process simulation environments used for nuclear power plant models, namely APROS. Next step is to develop first BioNuclear concept where nuclear heat and steam is used in a bio-refinery in order to produce bio-based chemicals and fuels from biomass.

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REFERENCES

Betova, I., Bojinov, M., Kinnunen, P., Lehtovuori, V., Peltonen, S., Penttilä, S. and Saario, T. 2006. In-situ measurements of the oxidation of AISI 316L(NG) and its constituents (Fe, Cr, Ni) in ultrasupercritical water. In: Proceedings of the 2006 international congress on advances in nuclear power plants - ICAPP'06. Reno, Nevada, USA. 4.-8.6.2006.

Galperin, A., Reichert, P. and Radkowsky, A. 1997. Thorium fuel for light water reactors - reducing proliferation potential of nuclear power fuel cycle. Science & Global Security. Vol. 6:3. P. 265-290.

Hwang, S., Lee, B., Kim, J. and Jang, J. 2008. SCC and corrosion evaluations of the F/M steels for a supercritical water reactor. Journal of nuclear materials. Vol. 372:2-3. P. 177-181.

Jalali, P., Li, M., Ritvanen, J. and Sarkomaa, P. Intermittency of energy in rapid granular shear flows. 2003. Chaos. Vol. 13:2. P. 434-443.

Jalali, P., Ritvanen, J. and Sarkomaa, P. 2006. Stress fluctuations in monodisperse and bidisperse rapid granular shear flows. Physica A. Vol 369:2. P. 535-544.

Kang, J. and von Hippel, F. N. 2001. U-232 and the Proliferation-Resistance of U-233 in Spent Fuel. Science & Global Security. Vol. 9:1. P. 1-32.

Kangas, P. 2009. Multi-Phase Chemistry in Process Simulation. Licentiate's thesis. Helsinki University of Technology.

Koster, A., Matzner, H. D. and Nicholsi, D. R. 2003. PBMR design for the future. Nuclear Engineering and Design. Vol. 222:2-3. P. 231-245.

Kurki, J. and Seppälä, M. 2009. Thermal Hydraulic Transient Analysis of the High Performance Light Water Reactor Using APROS and SMABRE. This conference.

Lee, J., Kimura, A., Kasada, R., Iwata, N., Kishimoto, H., Zhang, C., Isselin, J., Dou, P., Muthukumar, N., Okuda, T., Inoue, M., Ukai, S., Ohnuki, S., Fujisawa, T. and Abe, T. 2009. Super ODS Steels R&D for Fuel Cladding of Next Generation Nuclear Systems: 6) Corrosion Behavior in SCPW. In: Proceedings of the 2009 international congress on advances in nuclear power plants - ICAPP '09. Tokyo, Japan. 10.-14.5.2009.

Leppänen, J. 2007a. Development of a New Monte Carlo Reactor Physics Code. Espoo: VTT Technical Research Centre of Finland. 228 p. + app. 8p. VTT Publications 640. ISBN 978-951-38-7018-8.

Leppänen, J. 2007b. Randomly dispersed particle fuel model in the PSG Monte Carlo neutron transport code. In: Joint International Topical Meeting on Mathematics & Computation and Supercomputing in Nuclear Applications (M&C + SNA 2007). Monterey, California, USA. 15.-19.4.2007.

Moilanen, P. 2004. Pneumatic servo-controlled material testing device capable of operating at high temperature water and irradiation conditions. Espoo: VTT Technical Research Centre of Finland. 154 p. VTT Publications 532. ISBN 951-38-6384-0.

Motoe, S. and Hiroaki, S. 2006. Light Water Reactor Fuel Analysis Code FEMAXI-6 (Ver.1) - Detailed Structure and User's Manual. JAEA-Data/Code 2005-003.

Peng, Q., Teysseyre, S., Andresen, P. and Was, G. 2007. Stress corrosion crack growth in type 316 stainless steel in supercritical water. Corrosion. Vol. 63:11. P. 1033-1041.

Penttilä, S., Toivonen, A. and Heikinheimo, L. 2009a. Generation IV material issues – Case SCWR. This conference.

Penttilä, S., Toivonen, A., Heikinheimo, L. and Novotny, R. 2009b. SCC properties and oxidation behaviour of austenitic alloys at supercritical water conditions. In: 4th International Symposium on Supercritical Water-Cooled Reactors. Heidelberg, Germany. 8.-11.3.2009.

Penttilä, S., Toivonen, A., Heikinheimo, L. and Novotny, R. 2008. Corrosion studies of candidate materials for European HPLWR. In: Proceedings of the 2008 international congress on advances in nuclear power plants - ICAPP ’08. Anaheim, California, USA. 8.-12.6.2008.

Saito, N., Tsuchiya, Y., Kano, F., Ookawa, M., Kaneda, J., Moriya, K. and Matsui, H. 2006. SCC properties of candidate alloys for SCWR core components. In: Proceedings of the 2006 international congress on advances in nuclear power plants - ICAPP'06. Reno, Nevada, USA. 4.-8.6.2006.

Suikkanen, H. 2008. Coolant Flow and Heat Transfer in Pebble-Bed Reactor Core. Master’s Thesis. Lappeenranta University of Technology.

Zhang, Z., Wu, Z., Wang, D., Xu, Y., Sun, Y., Li, F. and Dong, Y. 2009. Current status and technical description of Chinese 2 x 50 MW_{th} HTR-PM demonstration plant. Nuclear Engineering and Design. Vol. 239:7. P. 1212-1219.