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NOMAGE4 activities 2011, Part I, Nordic Nuclear Materials Forum for Generation IV Reactors: Status and activities in 2011

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Abstract

A network for materials issues has been initiated in 2009 within the Nordic countries. The original objectives of the Generation IV Nordic Nuclear Materials Forum (NOMAGE4) were to form the basis of a sustainable forum for Gen-IV issues, especially focusing on fuels, cladding, structural materials and coolant interaction. Over the last years, other issues such as reactor physics, thermal hydraulics, safety and waste have gained in importance (within the network) and therefore the scope of the forum has been enlarged and a more appropriate and more general name, NORDIC-GEN4, has been chosen for the forum. Further, the interaction with non-Nordic countries (such as The Netherlands (JRC, NRG) and Czech Republic (CVR)) will be increased.
Within the NOMAGE4 project, a seminar was organized by IFE-Halden during 30 November – 1 November 2011. The seminar attracted 65 participants from 12 countries. The seminar provided a forum for exchange of information, discussion on future research reactor needs and networking of experts on Generation IV reactor concepts. The participants could also visit the Halden reactor site and the workshop.

Key words

NOMAGE4, Nordic-Gen4, seminar, Halden

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NOMAGE4 activities 2011, Part I

Nordic Nuclear Materials Forum for Generation IV Reactors:

Status and activities in 2011

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1. INTRODUCTION

Generation IV reactors (Gen IV) are a set of theoretical nuclear reactor designs currently being researched. Most of these designs are generally not expected to be available for commercial construction before 2030. Demonstration of concepts are however foreseen around 2015-2020. Current reactors in operation around the world are generally considered second- or third-generation systems, with the first-generation systems having been retired some time ago. Relative to current nuclear power plant technology, the claimed benefits for 4th generation reactors include 1) sustainability, 2) increased safety, 3) better use of the nuclear fuel, 4) high efficiency and better economics, 5) minimal waste production, 6) the ability to consume existing nuclear waste in the production of electricity, 7) increased proliferation resistance.

Many reactor types were considered initially; however, the list was downsized to focus on the most promising technologies and those that could most likely meet the goals of the Gen IV initiative. At present one focuses on six different types of reactors: 1) very-high-temperature reactor (VHTR), 2) supercritical-water-cooled reactor (SCWR), 3) molten salt reactor (MSR), 4) gas-cooled fast reactor (GFR), 5) sodium cooled fast reactor (SFR), 6) lead-cooled fast reactor (LFR). The VHTR has an open fuel cycle, the SCWR can have an open or closed fuel cycle, while the other concepts have closed fuel cycles. Because of their high operational temperature, several of the above concepts (GFR, MSR, VHTR) can be used for the cogeneration of hydrogen. Internationally, research is organized through the Generation IV International Forum (GIF), under the leadership of the US. The participation of Europe in GIF is through EURATOM, concentrating on the following concepts; GFR, SCWR, SFR and VHTR. Research into Generation IV reactors is also part of the activities of the European Sustainable Nuclear Energy Technology Platform (SNETP) which aims at reducing the green-house gas emissions by 20 % by 2020 (compared to 1990), make 20% energy savings and include 20 % share of renewable energies in the total energy mix.

VHTR In the past (70's and 80's), the High Temperature Reactor (HTR) concept has already proved its viability in US, GB and Germany operating up to 950 °C. The VHTR represents a modern and highly evolved version of the original HTR design. To improve in a consistent manner the economic performances, e.g. thermodynamic cycle and hydrogen processes efficiency, this new concept must produce an outlet gas temperature above 1000 °C. In order to fulfil these two major requirements (higher operating temperature and hydrogen production in terms of technological challenges and safety), an important international R&D program is underway to establish the viability of the VHTR by 2010 and to optimize its design features as well as operating parameters by 2015.

SCWR: The SCWR enables a thermal efficiency about one-third higher than current light-water reactors, as well as a simplification in the plant because the supercritical water can directly be used to drive the turbines (the steam conversion process is eliminated). The SCWR is a concept that uses supercritical water as the working fluid. SCWRs are basically light water reactors (LWR) operating at higher pressure and temperatures (500-600 °C). The SCWR is built upon two proven technologies, Light Water Reactors (LWRs), which are the most commonly deployed power generating reactors in the world, and supercritical fossil fuel fired boilers, a large number of which are also in use around the world.

MSR: Molten salt reactor systems (MSR) use liquid salts as a coolant and a fuel together. The system has a coolant outlet temperature in the range 700-800 °C, affording improved thermal efficiency. The main benefits of the MSR system are that it offers an integrated fuel cycle, embodying a burner/breeder reactor concept whilst taking advantage of the excellent heat transport properties of molten salt.

GFR: The high outlet temperature of the helium coolant used in the GFR system makes it possible to deliver electricity, hydrogen, or process heat with high efficiency. The outlet temperature would be of the order of 850 °C. The GFR uses a direct-cycle helium turbine for electricity generation, or can optionally use its process heat for thermochemical production of hydrogen. Through the combination of a fast spectrum and full recycle of actinides, the GFR minimizes the production of long-lived radioactive waste.

SFR: The SFR reactor concept is cooled by liquid sodium (at 550 °C) and fuelled by a metallic alloy of uranium and plutonium. The SFR's fast spectrum makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal spectrum reactors with once-through fuel cycles. One of the design challenges of an SFR is the risk of handling Sodium, which reacts explosively if it comes into contact with water.

LFR: The lead-cooled fast reactor features a fast-neutron-spectrum lead or lead/bismuth eutectic (LBE) liquid-metal-cooled reactor with a closed fuel cycle. The LFR is cooled by natural convection with a reactor outlet coolant temperature of 550 °C, possibly ranging up to 800 °C with advanced materials. Long-term R&D is required to develop high-temperature corrosion resistant materials.

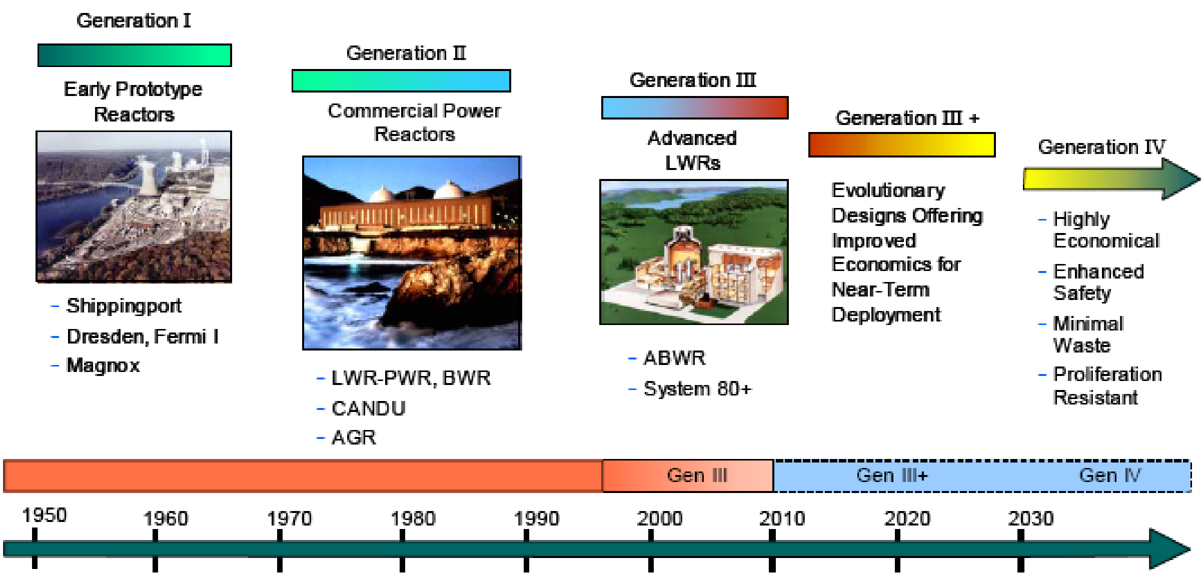


Figure 1: Time evolution of the different generations of reactor concepts.

Potential cladding and structural materials for the different GenIV systems are presented in Table 1.

Table 1 : Potential cladding and structural materials for the different GenIV systems [1-8]

System	Materials Cladding	 Core regions	 Out of core regions
Gas-Cooled Fast Reactor System (GFR)	Ceramics Matrices of SiC, ZrC & TiN, ODS (Oxide Dispersion-Strengthened)	Ceramics Carbides SiC, ZrC Nitrides ZrN, TiN Oxides MgO $ZrYO_2$ Zr_3Si_2 ODS	Coated or non coated ferritic-martensitic or austenitic steels Nickel based super alloys ODS
Lead-Cooled Fast Reactor System (LFR)	Austenitic, ferritic-martensitic steel Coated cladding e.g. FeAl		
Molten Salt Reactor System (MSR)	-	Graphite Nickel-based alloys Ceramics	
Sodium-Cooled Fast Reactor System (SFR)	ODS (Metallic fuel) Ferritic-Martensitic steels (MOX-fuel)	Advanced austenitic steels	
Supercritical –Water-Cooled System (SCWR)	Austenitic, Ferritic-Martensitic steels ODS		Ferritic- Martensitic steels
Very-High-Temperature Reactor System (VHTR)	ZrC	Graphite Ceramics	Ni-Cr-W super alloys High temperature metal alloys

There is a strong link between the materials integrity and nuclear safety for all nuclear reactors. The operation conditions in new generation reactors will be more demanding and thus knowledge about materials behaviour and integrity under operation are critical. To evaluate and choose proper materials to be used for GenIV reactors, it is important to know the boundary limits for use of the materials under the specific operating conditions. Safety analysis is based on technological assessment where the materials integrity has a decisive role.

2. OBJECTIVES OF THE WORK

The original aim of the project (in 2009) was to build a network for material issues related to Gen IV nuclear reactor systems. Since then, the network has evolved to include also other aspects related to Gen IV issues (fuels, reactor physics, thermal hydraulics, waste, safety, etc.). The idea of the network is to combine the resources of different research teams in order to carry out more ambitious and extensive research programs than would be possible for the individual teams. The network is focusing on the following aspects:

- Building of a sustainable Nordic network on GenIV issues, especially focussing on nuclear materials integrity and safety: fuels, cladding, structural materials and coolant interaction.
- Spreading knowledge on GenIV issues and on-going research to different parties involved in the nuclear field, such as LWR licensees universities.
- Inspiring young people to GenIV related R&D.
- Building a strong Nordic forum for GenIV collaboration.
- The results will be used to enhance new project ideas for international programs, e.g., Euratom FP7, calls with a strong Nordic partnership.

3. DELIVERABLES FOR 2011

A seminar (NOMAGE4) has been organized (30 October – 1 November) in Halden.

A new website has been initiated (www.nordic-gen4.org) which will allow for more possibilities in the future.

At IFE-Halden, instrument development for generation IV reactor research has been pursued. Significant progress has been made in the fabrication of instruments (such as the Linear Variable Displacement Transducer or LVDT) which can operate at even higher temperature (range 700 °C to 900°C) and which can also operate in liquid metals (sodium or lead). Further, coatings have been tested (out of pile and in-pile) for protection against corrosion and for reducing the diffusion of hydrogen and tritium through metals.

The cooperation between the Nordic institutes, companies and universities has been strengthened.

The company GE-Hitachi Nuclear (Sweden) has become member of the network in 2011.

The Joint Research Centre (JRC, Institute for Energy and Transport (IET), Petten, The Netherlands), Nuclear Services for Energy, Environment & Health (NRG, Petten, The Netherlands) and the Research centre Rez (VCR, Rez, Czech Republic) also became members in 2011 and will become partners in 2012.

4. ORGANIZATIONS INVOLVED IN NOMAGE4

The partners are presently:

Table 2: Partners of NOMAGE4 in 2011

1	Institutt for Energiteknikk (IFE/Halden Reactor Project)	www.ife.no
2	Studsvik Nuclear AB	www.studsvik.com
3	VTT	www.vtt.fi
4	Risø DTU	www.risoe.dtu.dk

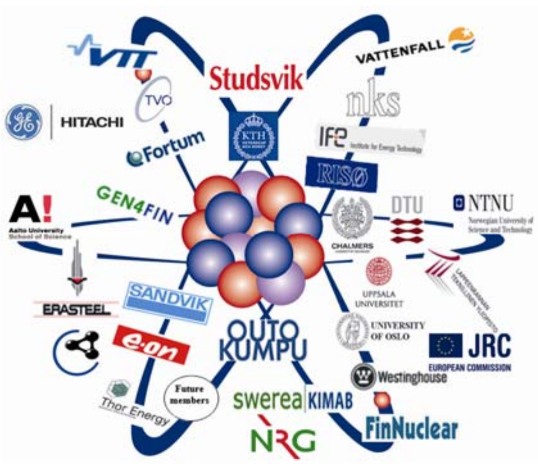


Figure 2: Present Logo for the network

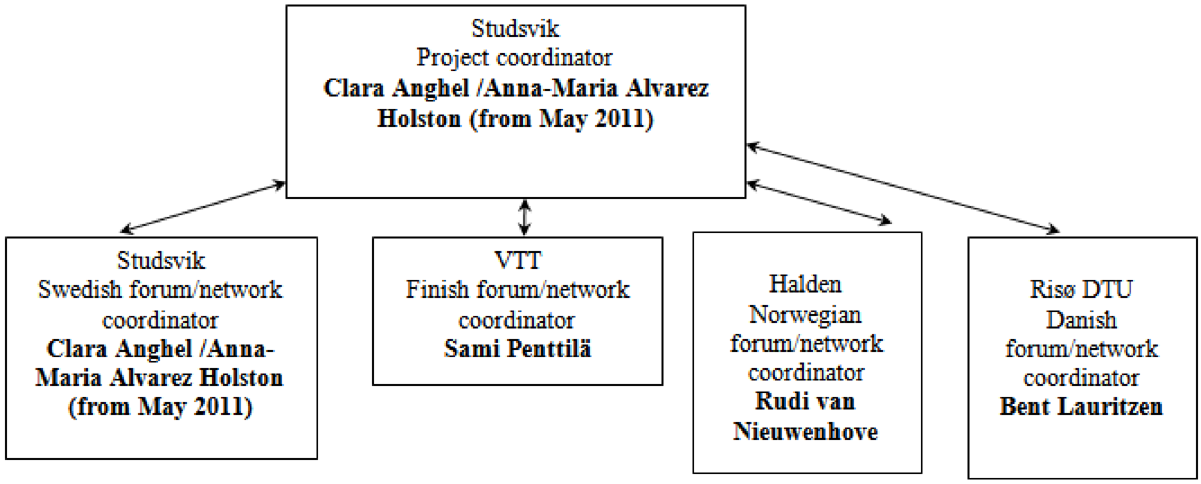
The members (including partners) are:

Table 3: Members of the NOMAG4 network

1	Institutt for Energiteknikk (IFE)	www.ife.no
2	Studsvik Nuclear AB	www.studsvik.com
3	VTT	www.vtt.fi
4	Risø DTU	www.risoe.dtu.dk
5	Vattenfall	www.vattenfall.se
6	Fortum	www.fortum.com
7	TVO	www.tvo.fi
8	Westinghouse	www.westinghousenuclear.com
9	EON	www.eon.se
10	Sandvik	www.sandvik.com
11	Outokumpu	www.outokumpu.com
12	Lappeenranta University of Technology	www.lut.fi
13	Aalto University	www.aalto.fi
14	GEN4FIN	http://virtual.vtt.fi/virtual/gen4fin/
15	Royal Institute of Technology, KTH	www.kth.se
16	Chalmers University of Technology	www.chalmers.se
17	Uppsala University	www.uu.se
18	Thor Energy, Norway	www.thorenergy.no
19	GE-Hitachi Nuclear Energy	www.ge.com/nuclear
20	Swerea KIMAB	www.swerea.se
21	FinNuclear Prizztech	www.prizz.fi
22	University of Oslo	www.uio.no www.mn.uio.no/kjemi/forskning/grupper/safe/
23	JRC: Joint Research Centre (Petten, The Netherlands)	http://iet.jrc.ec.europa.eu/
24	NRG: Nuclear Services for Energy, Environment & Health (Petten, The Netherlands)	www.nrg.eu/

25	Research Centre Rez	www.cvrez.cz/web/en

The structure of the Nordic-Gen4 network (in 2011) was as follows:



From 2012, the structure will be as follows:

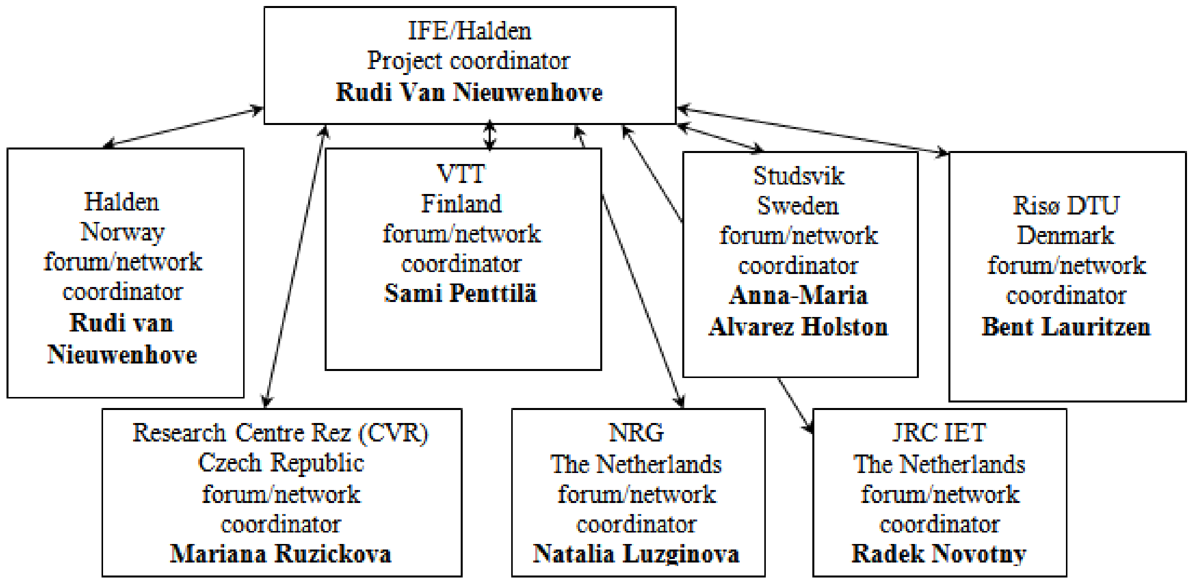


Figure 3: Schematic presentation of the activity organisation for the project NORDIC-GEN4.

5. NOMAGE4 SEMINAR IN HALDEN

A NOMAGE4 seminar (2 days) was organized by IFE/Halden (by Rudi Van Nieuwenhove and Katrine Fuglesang Øhre). The seminar attracted 65 participants from 12 countries. In addition to the 4 Nordic countries, there were participants from Belgium, The Netherlands, France, Germany, United Kingdom, Canada and Czech Republic and Estonia. The participants had the occasion to visit the reactor site and the workshop. The programme can be found in Appendix A. Based on the reactions during and after the seminar, it can be concluded that the participants were very satisfied. The seminar was funded by NKS and sponsored by Halden, Studsvik and GE-Hitachi Nuclear Energy.

5.1 Summary of the oral and poster presentations

October 31st 2011

Margaret McGrath, Deputy Project Manager of the Halden Project and Research Director of the sector Nuclear Safety and Reliability, gave a general introduction to IFE and the Halden Reactor Project.

Fridtjov Øwre, General Manager of the Halden Project, presented HUNISEC (Halden-Universities-Secondee program). Employees from member organizations can stay and work at Halden for a period of 1 – 2 years (Secondee arrangement). Collaboration with various universities has been handled up to now case by case. HUNISEC is directed at establishing a more formal arrangement between member organizations, universities and the HRP that might lead to a win-win situation for the three parties.

The main aims of HUNISEC are to:

- Increase knowledge transfer from Halden to member countries.
- Increase the number of Secondees by inviting Master students, PhD's and Post-Docs to the HRP to work with their thesis in IFE's experimental environments
- Provide opportunities for member country students to study and work abroad in a small but renowned international research institute
- Provide opportunities for the HRP to access fundamental methods, techniques and tools available from various member Universities for use in applied research at the HRP

Fazio Concetta (KIT, Germany), coordinator of the European program GETMAT and overall coordinator of the Joint Program on Nuclear Materials (JPNM) of the European Energy Research Alliance (EERA), presented SNETP, EERA (objectives, organization, programmes), JPNM and the programs GETMAT and MATTER. Within the European Sustainable Nuclear Industrial Initiative (ESNII), three Gen-IV concepts are pursued; 1) Sodium Cooled Fast Reactor (SFR) as reference technology and the Lead Cooled Fast Reactor (LFR) and the Gas Cooled Fast Reactor (GFR) as alternative technologies.

Within JPNM, 4 sub-programmes have been defined: 1) Support to the ESNII, 2) Oxide Dispersed Strengthened steels (ODS), 3) Refractory materials; Ceramic composites and metal-based alloys, 4) Modelling: Correlation, simulation and experimental validation. The first EC supported pilot project for JPNM is MATTER. Within MATTER, test and evaluation guidelines for structural materials (related to MYRRHA and ASTRID) will be developed and pre-normative R&D for codes and standards will be performed. The SET-plan Materials Road Map initiative has allowed identifying synergies with other energy technologies.

Rudi Van Nieuwenhove (IFE, Halden), presented the objectives and structure of the NOMAGE4 network. From 2012, Halden will take over the coordination of the network. The network will be extended to include also some non-Nordic participants (JRC, NRG, VCR). In addition, the website will be changed to www.nordic-gen4.org.

Annika Olsson (Ångström Materials Academy (ÅMA), Uppsala University), explained the role and objectives of ÅMA. ÅMA is an interface between the materials science research at the Ångström Laboratory and the industry. During the last year, the ÅMA board wants to organize meetings for academy and industry around Gen-IV. Some of the companies involved are; Vattenfall, Sandvik, ABB, Outokumpu, OKG AB (Oskarshamns Kraftgrupp AB, part of the E.ON group) and Uddeholm (steel producer).

Peter Szakalos, KTH, **Ane Håkansson,** Uppsala University and **Christian Ekberg**, Chalmers University of Technology gave an overview (combined talk) of the GENIUS project (see http://genius.kth.se). GENIUS is a science council funded collaboration between KTH, Chalmers and Uppsala University, aiming at developing technology necessary for safe and economic deployment of Generation IV nuclear reactors, in particular the lead cooled fast reactor. The coordinator is Janne Wallenius professor in reactor physics at KTH. The project is organized in three major focus areas: Fuel development, Materials research, Safety and security. In the fuel work package of GENIUS a national laboratory for development of advanced Generation IV fuels has been set up and is partially in operation since 2011. The laboratory will permit fabrication of solid solution nitride fuels as well as composite oxide fuels.

Hamid Aït Abderrahim, Deputy Director-General of SCK-CEN and Project Director of MYRRHA, presented the MYRRHA project, the international collaborations and the MYRRHA International Consortium. MYRRHA (see www.myrrha.be) is an Accelerator Driven System (ADS) in which a high energy proton beam is coupled to a subcritical core, resulting in a significant fast neutron source. MYRRHA will be a multipurpose facility for fuel and materials research, for production of radio-isotopes, silicon doping, fundamental research, fusion and waste. Using transmutation, it is possible to obtain a factor 1000 reduction in the required storage time for nuclear waste. Belgium has committed 40 % of the required investment, while the remaining 60 % has to come from an international consortium. The total cost is estimated to be EUR 960 million. Full operation is expected to be from 2023.

Milan Tesinsky (KTH, Sweden) gave a presentation on the impact of americium (Am) on the transient analysis of the European Lead System (ELSY). From September 2006 to March 2010, the ELSY project developed a very innovative pre-conceptual design of an industrial plant for electricity production that can close the fuel cycle. The LEADER project, financed in the frame of EU-FP7, started on April 2010 from the results achieved by ELSY with an in-depth analysis of the actual reactor configuration.
The effect of Am on the effective delayed neutron fraction, the Doppler constant and the coolant void worth were investigated by means of the Monte Carlo SERPENT. The following transients were considered: Unprotected Transient Over Power and Unprotected Loss of Flow. Peak fuel temperature and peak cladding temperatures increase significantly with increasing Am content. It was concluded that the reference design of ELSY was not suitable for Am recycling. It was pointed out however that this problem could be solved by the use of nitride fuel.

Youpeng Zhang (KTH, Sweden) gave a presentation on the transmutation of Americium in Sodium Cooled Fast reactors. Americium is a troublesome element as 1) it has a high radiotoxicity, 2) it is long-lived and 3) it is non-transmutable in a LWR. Its presence causes also difficulties in spent fuel handling. Using a fast spectrum, it is possible to transmute Am. SERPENT model calculations were performed for three cases: 1) BN600 reactor (in operation), (U, Pu, Am)O_2 fuel, 2) IFR reactor (U, Pu, Am)Zr fuel and 3) BN1200 reactor (under construction), (U, Pu, Am) fuel. It was concluded BN1200, loaded with nitride fuel is the most efficient facility to transmute americium.

Sten Anders Wilson, (Sandvik, Sweden) gave a general presentation on Sandvik's products. The Sandvik Group consists of 3 business areas: Tooling, Mining and Construction and Materials Technology and the talk concentrated on products by Sandvik Materials Technology: Steam generator tubes, fuel cladding tubes, tube and pipe for nuclear power applications, wire, strip and bar products for nuclear fuel components and welding materials. Materials of interest are; alloy 690, alloy X-750, alloy 718, alloy 800, Zr2, Zr4, Zirlo, 316L, 12R72(15Cr15NiMoTi), 2.3Cr1MoNiNb and 9Cr1Mo.

Wenyue Zheng (CanmetMATERIALs, NRCan, Canada) gave an overview of the research in Canada on materials for use in supercritical water in relation to the CANDU-SCWR concept. Many new facilities for materials R&D exist, or are being set up. Presently, there are 5 SCWR Material Research Projects, 8 SCWR Chemistry Research Projects, 5 SCWR Thermal Hydraulic Research Projects and 5 SCWR Research Safety Projects. The CANDU-SCWR will retain the main CANDU features (modular fuel channels, heavy water moderator) but will operate with supercritical water at a pressure of 25 MPa and an outlet temperature up to 625 °C. Stress corrosion cracking was studied in SCW on SS316, SS310 and Inc 625. The smallest crack depths were observed in Inc 625 and the largest in SS316. It was also found that (collaboration between MTL, AECL-CRL, VTT) that there is a significant effect of grain size on Stress Corrosion Cracking (SCC) susceptibility, with the lowest corrosion rate for the fine grain alloys. The effect of oxygen on corrosion was studied in various alloys, including ODS steels. The second phase of the Canadian program focuses on: 1) Research necessary to develop a viable fuel-channel design, 2) GIF SCWR Fuel Qualification Loop Project, inc. materials R&D, 3) Develop key technical solutions needed to implement the Canadian SCWR concept.

1st November 2011

Jesper Ejenstam (KTH, Sweden) presented an evaluation of experimental FeCrAl alloys in liquid Pb. Novel experimental FeCrAl alloys, with 10 wt.% Cr and 4-8 wt.% Al, were exposed to liquid lead, containing 0,1 wt. PPM O, at 550°C for 2500 h. The study aimed at evaluating the corrosion resistance of the experimental FeCrAl alloys, and to compare them with commercial FeCrAl alloys. The chromium content of the experimental alloys were lowered in order to try to make them resistant to spinodal decomposition, a embrittlement phenomenon that occurs at about 475°C, which is about the same temperature as the operating temperature of a lead cooled fast reactor. The commercial alloy, Kanthal APMT (22 wt.% Cr and 5 wt.% Al), and experimental Kanthal AF (21 wt.% Cr and 5,3 wt.% Al) showed good performance with thin comprehensive aluminium rich oxides formed on the steel surface. All experimental alloys showed deteriorated oxidation properties in comparison with the commercial alloys, mainly due to the lower chromium content. However, it could be seen that higher aluminium content was beneficial for the oxidation properties, with less internal oxidation. The effect of small additions of rare earth metals (REM) was also shown, and the alloy without addition of REM (10 wt.% Cr and 6 wt.% Al) showed poor oxidation properties with an inward growing spinel and an outward growing magnetite scale. It should be mentioned that none of the novel experimental FeCrAl alloys were optimized in REM additions, which is crucial for the oxidation properties. However, in comparison with 15/15 Ti, an austenitic reference material for nuclear applications, all of the experimental alloys showed good performance regarding oxidation properties. The experimental alloys showed oxide thicknesses in the range of 1-25 μm, whereas the austenitic 15/15 Ti show an oxide thickness of about 25 μm in the same environment.

Serguei Gavrilov (SCK-CEN, Belgium) discussed the material issues for design and licensing of the MYRRHA ADS system. The following topics were discussed: 1) The role of materials research for MYRRHA development, 2) Challenges, 3) The Myrrha materials program. Some important material data (needed for the design) are still missing, such as basic characteristics of candidate materials (T91, SS 316L, 15-15Ti), the effects of Liquid Metal Embrittlement (LME) on materials properties, SCC, et. It was also pointed out that the existing material properties in codes are derived from the results of multiple tests performed on multiple material heats and that the “distance” between single experimental data points and the “material property for design” is often quite large. In addition, many existing experimental data are inconsistent, probably because some important parameters were not reported. Because of the limited experimental capacity (and the limited number of liquid metal fast reactors), absence of accepted test procedures, big scatter of test results and the limited understanding of the underlying phenomena, wide safety margins have to be applied for the design. The MYRRHA materials R&D program was explained in more detail. One of the important issues mentioned were the development of testing procedures, which are dealt with through the FP7 MATTER project. Another important issue is the development of the required testing infrastructure.

Rudi Van Nieuwenhove, (HRP, IFE) presented an overview of the activities of the Halden Reactor Project (HRP) into research for Generation IV reactors. IFE/HRP, in collaboration with Risø DTU, performed a detailed study on an in-pile SCW loop in the Halden reactor. As no funding could be obtained from the Norwegian Research Council (Norske Forskningsrådet), HRP will now try to find external partners for co-funding this loop. The many advantages of the SCW reactor concept were pointed out: 1) No need for steam generator, 2) Many years of experience with supercritical fossil-fuel fired power plants, 3) Compact design, 4) Water as coolant (simplifying inspection), 5) Low coolant mass flow rates, 6) Less susceptible to earth quakes, 7) Corrosion can be managed, 8) Existing materials and fuels can be used. Further, the myth of the SCWR being a thermal reactor (by definition) was debunked. Reference was made to fast SCWR concepts developed in Japan and China. These fast reactors are also capable of breeding fuel and burning actinides. Therefore, it was argued that the SCWR should not have been left out of the ESNII /SNETP road maps and that this point should be reconsidered.

Instrument development for Gen-IV at the HRP focuses on the development of very high temperature LVDTs, development of electrochemical potential sensors and crack growth measurements in liquid metal. At present, HRP is the only producer of ECPs which can operate in supercritical water.

Various coatings are under investigations: TiAlN, CrN and ZrO_2. Coating by CrN was shown to be well suited for SCW, while coating by ZrO_2 is best suited for use in liquid lead. In-pile tests on instruments and coatings are on-going in the Halden reactor.

Thomas Schulenberg (KIT, Germany) presented the plans for in-pile testing of a small scale fuel assembly under supercritical water conditions. The out-pile SCW-loop, presently in operation in Řež was described. It is planned to install this loop in the LVR-15 reactor at Řež such that in-pile tests of a fuel assembly under SCW conditions become possible. Out of pile validation tests of a test section with 4 electrically heated fuel rods will be performed at Shanhai Jiaotong University in China in the supercritical water loop SWAMUP. For the in-pile tests in LVR-15, it is planned to load 4 fuel rods. The conditions (temperature, pressure, linear heat rate) for the in-pile tests are based on the predicted conditions in the HPLWR reactor concept. The aim of the European project SCWR-FQT (which runs from January 2011 to December 2013) is to license the supercritical water loop at Řež as a nuclear facility. In this project, a test facility will be analysed under normal and accident conditions. An emergency coolant line has been included for cooling the fuel in case of pressure loss in the loop. The material options for the cladding material presently considered and studied are 1.4970, TP347H and 316L.

Sami Penttilâ (VTT, Finland) gave a presentation on the effect of surface modification on the corrosion resistance of 316L in supercritical water conditions. This work was made in co-operation with CANMET, Canada. One of the critical technological issues in SCWR is the materials for fuel cladding and core components and at this point, no single alloy has received enough study to ensure its long-term performance under SCW conditions up to 650 °C. Results were presented on the impact of the surface condition (as-machined, or ground with papers of different grit sizes (600 and 1200)) on corrosion of 316L. Tests were carried out in the SCW-loop at VTT at 650 °C and 250 bar with exposure times between 1000 h and 3000 h. The samples were analysed by means of weight gain, Scanning Electron Microscopy (SEM), Energy Dispersive X-ray Spectroscopy (EDS), Focused Ion Beam Microscopy (FIB) and Transmission Electron Microscopy (TEM). Best corrosion resistance was observed on the as-machined samples. The protectiveness of the thin protective oxide films extended at least up to 3000 h. The thin oxide film had a maximum chromium concentration of about 50 %. This high chromium content was attributed to outward diffusion (from the base material: 316L) through a fine-grained re-crystallized layer.

Marketa Zychová (Research centre Rez, Czech Republic) presented the water chemistry for SCW. First, an overview was given of the water chemistry used in fossil-fuelled supercritical water-cooled power plants. A fossil-fuelled (FF) SCW plant is presently under construction, namely Ledvice 6 Power Plant. By using the proper water chemistry, it is possible to reduce corrosion product transport and release, to optimize the thermal performance and to maximize component life. In FF SCW plants, one uses a combination of ammonia and hydrazine but also chelate is used to increase the thermal conductivity of iron oxide deposits. The selected water regimes for the SCW reactor are presently those based on the boiling water reactor (BWR), namely normal water chemistry (NWC) or hydrogen water chemistry (HWC) in with a certain amount of dissolved hydrogen. For the SCWR, the main problem is however the great difference between input and outlet temperature. This changes for instance the solubility of gases and corrosion products. It is important to reduce contamination by corrosion products for the good operation of the turbines. This is important as no recirculation loops are foreseen to provide purification and treatment of the medium. Next, the facilities at the Research Centre Rez (part of the Nuclear Research Institute Group (NRI)) were discussed. The SCW-loop set-up was explained in more detail. At present, the loop is located in a separate building for out-pile testing. It is planned to install this loop in the LVR-15 reactor.

Radek Novotny (JRC, The Netherlands) gave a presentation on corrosion and SCC material testing at JRC IE Materials research at JRC is performed in the framework of the Mattino project (Material performance for innovative reactor systems) which includes studies for the HPLWR and the FQT SCWR EU projects. JRC has several autoclaves for SCW and a SCW loop for materials testing. Testing is performed by the potential drop techniques (DCPD), acoustic emission and electrochemical methods. Whereas most Stress Corrosion Cracking (SCC) tests have been conducted using Slow Strain Rate Testing (SSRT), a bellows loading test system has been developed at JRC. This system has several advantages such as reduced size and that there is no need for movable parts going through the top seal. Precracked Charpy type specimens have been tested. Results on AISI 421 (8Cr18Ni10Ti) were shown. Crack growth rates in SCW were found to be significantly lower in SCW as compared to BWR. A new, double-bellows type of lading device has been developed in which one bellows is used to compensate for the environmental pressure.

David Powell (GE, Hitachi, UK), presented fuel recycling using the PRISM reactor concept. The Nuclear Fuel Recycling Centre (NFRC) produces PRISM fuel from the recycled uranium and long-lived isotopes. The short-lived isotopes are isolated into stable waste forms. PRISM is a recycling reactor which consumes waste and U235 and produces power (400 W/cm^3). A PRISM power block consists of two sodium-cooled reactors which drive one turbine (622 MWe output). Safety is guaranteed through passive decay heat removal and automated safety grade actions. Because it is a relatively small, simple and modular structure, capital and investment risks are reduced. PRISM can also operate as a Pu burner. This is an important aspect in view of the 82 tonnes of Pu presently stored at Sellafield (and expected to grow to 100 tons). Using Pu as fuel minimizes Pu transport and ensures proliferation resistance. PRISM is ready for deployment and is claimed to be the most advanced Gen-IV reactor and the waste storage needs are reduced significantly.

Larry Nelson (JLN Consulting, USA) could not attend the meeting due to problems with his flight. He sent however his presentation entitled “Advanced Radiation Resistant Materials (ARMM) Program”. Existing alloys used for LWR reactor internals have been known to degrade in less than 40 years. The operating life on the other hand is being extended to 60 years and operation beyond 80 years is under consideration. Therefore, component replacement may be required in some reactors. Existing alloys may not ensure economic operation to the end of the extended life because of IASCC, decrease in fracture toughness or swelling. Advanced materials will allow reactors to be operated with greater efficiency, characterized by lower maintenance, inspection and repair costs. The motivation for setting up the ARMM program is based on the fact that research on irradiated materials is often too costly for one organization to sponsor. The objectives of the ARMM program are: 1) By 2021, to develop and test a radiation resistant alloy within the current commercial alloy specifications, 2) By 2024, to develop and test a new advanced alloy with superior radiation resistance. A 10 year program, cooperatively sponsored, will start in 2012. The program workscope will be defined by sponsors and managed by EPRI.

Rune Hoel (MOTecH plasma, Norway) gave a presentation on Pulse Plasma Surface Treatment (PPST), a technique used for wear, corrosion and diffusion protection. This technique is performed at low temperatures (preserving core properties like toughness, creep, ductility) and could find applications in the nuclear industry, and more particularly to Gen-IV reactors. The technique is a plasma surface treatment and not a coating. It can be applied to very large surfaces of complex geometry and even the inside of pipes can be treated. Different types of materials can be treated such as carbon steels, low alloy steels, stainless steels, inconels and titanium alloys. The hardness of the materials can be improved significantly, thereby reducing galling and wear. For hydrogen diffusion studies, the work is carried out in collaboration with IFE.

Sunniva Rose (UiO,Norway) discussed the potential of using thorium in existing reactor designs. The motivation behind this is to economize the available uranium resources and solve (minimize) the waste problem. The waste production with Th is smaller because of the smaller capture to fission ratio of U233 (derived from Th232) is smaller for U233 as compared to Pu239 (derived from U238). In existing reactors, a self-sustained Th cycle is not possible and one still depends on the existing U/Pu cycle. In the future (using Gen-IV concepts), such a self-sustained Th cycle will however be possible. At IPN Orsay and LPSC Grenoble, MCNP codes have been developed (MURE) for reactor evolution calculations. It is a precision research code, well adapted to innovative systems and fuels and it is open source, available at NEA. A study (by S. Rose, J.N. Wilson, et. al.) to minimize actinide waste has been made based on multi-recycling of thorium. Using multi-recycling, a 50 % reduction in natural uranium needs can be obtained. Further, actinide cross sections were examined experimentally at OCL and SAFE (Oslo).

Jaap G. van der Laan (NRG, the Netherlands), presented the activity at NRG (at the HFR reactor at Petten) on testing of fuels and materials for next generation reactors. The three main pillars of the NRG Research & Development are 1) Fast Breeder Reactors and P&T, 2) LWR (SCWR), (V)HTR cogeneration, being in line with the European nuclear R&D strategy as laid down by the SNETP. R&D work at NRG is either funded commercially, 50%-50% share cost by EU and NRG via European framework programs, or by NRG internal budgets. NRG is participating in several European Framework programs (ACSEPT, ADRIANA, ANDES, ESFR, GOFASTR, LEADER, CDT, FAIRFUELS, F-BRIDGE, GETMAT, MATTER) and now also PELGRIMM, ASGARD and SEARCH. A major challenge for transmutation fuels is helium production as this leads to fuel swelling (and consequent fuel-cladding interactions) and He-release (increase in pin pressure). The BODEX project aims to investigate these effects by including boron to produce the helium. PIE is still on-going. Within the FAIRFUELS project, NRG conducts two irradiations in 2010-2012, namely MARIOS and SPHERE in which americium containing fuel will be investigated. Further, irradiations in liquid lead bismuth eutectic (LBE) are performed to examine the compatibility of 316L and 9Cr steels with LBE. Within MATTER, stress corrosion cracking is being examined on CT specimens (9Cr steel). Several studies regarding the VHTR are on-going and NRG is also involved in the ARCHER (Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity Research and Development), a new HTR European FP7 R&D

program. NRG is also doing research in the field of fusion (ITER, vessel, bolt materials, weldability of neutron irradiated SS-316L(N) steel, etc.).

Tor Bjørnstad (IFE, Norway) gave a presentation on the extraction of Th from Norwegian mineral resources. The available Th resources in Norway are estimated to be 132000 tonnes of Th metal. The many advantages of using Th were listed. Breeding in the Th232-U233 cycle can be obtained in thermal, epithermal and fast reactors. The fission product release rate from Th is a factor 10 less than that from UO2 and the disposal time from spent fuel is 500 years, as compared to 200000 to 500000 years for U based fuel. Significant amounts of the highly radioactive 232U and strong gamma-emitting decay products necessitates however remote and automated reprocessing and fuel fabrication.
The experience of Th fuels and fuel cycles is very limited as compared to UO2 and (U,Pu)O2. Further research and investigations are needed before massive investments into commercial utilization. The presence of thorium in the Fen area in Norway was explained in more detail. Most of the thorium there is in the form of thorianite and is present together with silicium, calcium, iron, phosphorous. Many physical and chemical methods of Th extraction from the ore were explained.

Anna-Maria Alvarez Holston (Studsvik, Sweden) described In-pile and Out-pile methods to predict fuel cladding failures. The effect of burn-up on fuel cladding, fission gas release (FGR) and pellet expansion was described in detail. Special attention was paid to pellet cladding interactions (PCI) leading to stress corrosion cracking (enhanced by fission products such as iodine). Hydrogen embrittlement and delayed hydride cracking are important, though they are not observed in the statistics. This can be due to the fact that, in visual inspection, the macroscopic features are almost identical to PCI failures, i.e. axial cracks. Up to 3000 wppm H, Zr-alloy claddings are ductile (at room temperature) but due to the temperature gradient in the cladding, the hydrogen concentration can peak at 9000 wppm in the outer rim, causing a very brittle structure. The importance of ramp testing was stressed because otherwise one can miss conditions that trigger certain failure mechanisms.

Barbara Oberländer (IFE-Kjeller, Norway) gave an overview of the material characterization capabilities at IFE Kjeller (NMAT). NMAT supports the JEEP II reactor and the Halden reactor with fuel and hot cell services. NMAT is responsible for PIE of nuclear materials and fuels, for the fabrication of fresh standard and experimental nuclear fuel, refabrication and instrumentation of pre-irradiated fuel, nuclear waste handling, intermediate storage and safeguarding. The PIE includes a wide range of non-destructive and destructive test methods. These methods are important also for the investigation of Gen-IV materials and fuels. Irradiated steels, zircaloy and fuels can for instance be analysed by an analytical SEM (on replicas or miniature specimens). Irradiated fuel can be examined visually, by metallography and ceramography and it is possible to perform hydrogen analysis and tensile testing. For the fabrication of test samples from irradiated materials (such as CTs, tensile specimens) a remotely handled milling machine in a hot cell is available. Irradiated inserts can also be welded on CT specimens for crack growth measurements (by means of the DCPD technique).

Posters

***Thorium resources in Norway and potential applications in Generation-IV reactors**,*
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Norway has one of the major thorium resources in the world, about 15%. The listed resources of 170 000 tonnes have a potential energy content which is about 50 times larger than all oil&gas extracted to date by Norway, plus that of the remaining reserves. The most promising resource of thorium enriched minerals are found in the Fen Complex in the Telemark County 120 km South West of Oslo with thorium amounting to about 0.1–0.4 wt% (weight per cent).This paper provides more details about the resources and proposes new research activities to determine more precisely the amount of thorium available and distribution of thorium in the rock minerals. One challenge identified in earlier studies is that the minerals are so fine-grained (less than 40 micrometers) which means that thorium cannot be extracted with satisfactory recovery using traditional techniques. Various combinations of physical and chemical methods as candidate technologies for thorium extraction in these rocks are discussed. The potential benefits of using thorium vary depending on the type of reactor considered. Focus is put on the Molten Salt Reactor (MSR), one of the Gen-IV options. The overall objective is to initiate activities that can provide more knowledge to further assess whether thorium in Norwegian rocks can be defined as an economical asset for the benefit of future generations.

Assessment of Feasibility of Thorium Fuel in BWRs

R. P. J. Vanhanen and P. A. Aarnio

Uranium fuels consist of fissile U235 nuclide, fertile U238 nuclide and bred fissile Pu239 nuclide. Thorium fuels exchange some of U238 for fertile Th232 which breeds fissile U233. The U233 nuclide has favourable neutronic properties compared to other fissile nuclides mentioned. We restrict the study to oxide fuels. First we compare two lattice calculation codes to assess their capability to predict the behavior of thorium fuel. Deterministic licensing level code CASMO-4E [1] is compared to a Monte Carlo code Serpent [2]. ENDF/B-VI based libraries are used for both codes, although no libraries with the same release were available. The main interest lies in the group constants and discontinuity factors which are used by higher level reactor analysis codes. The differences between results of the codes do not grow when uranium fuel is switched to thorium fuel. Second we design five new thorium fuel assemblies and compare them to a reference uranium fuelled assembly. Both high and low thorium content variations are considered. The reference assembly is a 10x10 BWR assembly with a 2x2 water tube in the middle. For simplicity, the assemblies have no axial variation. The thorium assemblies are designed so that their energy output will be approximately equivalent to the uranium assembly. The U235-U238-Th232 90Th mixture, pin pitch, pin radius and gadolinium content are varied in manual local optimization. Design objectives, compared to the reference assembly, are more negative reactivity feedbacks, lower local power peaking factor, lower 235 92U usage and steadier decline of the multiplication factor. The generated libraries of group constants are used in whole reactor simulation using advanced nodal code SIMULATE-3 [3]. The reactor analysis is simplified by considering only a single assembly type running an equilibrium cycle. Only current industry practice of once-through fuel cycle is considered. Parameters of interest include power peaking factors corresponding to thermal margins, discharge burnups, coefficients of reactivity and shutdown margin. None of the thorium assemblies could improve all parameters, but all designs improved on some aspects.

[1] Studsvik Scandpower, CASMO-4E, User's Manual — University Release, SSP-09/442-U Rev 0, 2009.

[2] Leppänen, J., Development of a New Monte Carlo Reactor Physics Code. Doctoral Thesis, Espoo, VTT Publications 640, 2007.

[3] Studsvik Scandpower, SIMULATE-3, User's Manual — University Release, SSP-09/447-U Rev 0, 2009.

Pulsed Plasma Surface treatment

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Pulsed plasma surface treatment (PPST) has been widely applied within the automotive- and tool industries for more than a decade. Such plasma assisted diffusion treatments in the form of nitriding, nitrocarburizing and oxidation have offered an environmentally friendly alternative to detrimental chemical processes. Metals which are traditionally difficult to surface-harden, e.g. NI-based alloys, may also have their structure and properties improved.

The PPST also has a potential as a cost-efficient, "green" process for applications within the nuclear industry. Various pulsed plasma processes may be applied to a wide variety of metals, ranging from simple carbon steels to stainless steels, Ni-based superalloys, and zirconium- and titanium alloys. PPST may improve the wear and corrosion properties of components, as well as producing diffusion barriers against detrimental atomic species (e.g. hydrogen).

Surfaces may be modified with excellent shape-, surface roughness- and size stability. Hence, there will be no need for costly mechanical- or thermal post-treatments. It is also relatively simple to modify the surfaces of complex geometries and to scale up the process for large components.

Simulating transmutation in MYRRHA with Fluke MC-code

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The MYRRHA [1] accelerator driven system (ADS) currently planned in Belgium, is meant to be a multipurpose irradiation facility for GEN IV and fusion reactor materials. Additionally, its purpose is to demonstrate and further develop the capabilities of ADS designs. MYRRHA will make use of a proton accelerator and a lead-bismuth spallation target to create neutrons for the multiplying subcritical core. It will use U-Pu MOX fuel and liquid lead-bismuth as coolant. In addition, since it is a fast spectrum system with a very controllable neutron source, it is ideal for studying transmutation of minor actinides and long lived fission products. In this work, the performance of the MYRRHA core is simulated with FLUKA [2,3[, which "is a fully integrated particle physics Monte Carlo simulation package [4].

In the simulation the full MYRRHA core geometry and materials were modelled together with the 600 MeV proton beam hitting the spallation target, which provides the fast neutrons to the multiplying core. The simulation showed the neutron yield to be 9.9 neutrons per primary proton and energy multiplication factor 22, with which the 3.5 mA proton beam provides a power of 76 MeWth. These values are consistent with the MYRRHA design. We used the irradiation positions of MYRRHA to transmute Am-241, I-129 and Tc-99. Preliminary simulations gave the following transmutation rates: 1.3E-5 per day for Am-241, 6.9E-6 per day for I-129, and 1.2E-5 per day for Tc-99. These rates are too small to have any practical consequences. Furthermore, the amount of material to be transmuted in one batch turned out to be limited because it very soon began to decrease the neutron flux. In this specific case better transmutation rates would have been obtained by using the primary proton beam directly for the irradiation. Further transmutation simulations in MYRRHA will be performed with critical core configuration.

1. http://myrrha.sckcen.be/
2. "The FLUKA code: Description and benchmarking", G. Battistoni, S. Muraro, P.R. Sala, F. Cerutti, A. Ferrari, S. Roesler, A. Fasso, J. Ranft, Proceedings of the Hadronic Shower Simulation Workshop 2006, Fermilab 6--8 September 2006, M. Albrow, R. Raja eds., AIP Conference Proceeding 896, 31-49, (2007).

3. “FLUKA: a multi-particle transport code” A. Fasso, A. Ferrari, J. Ranft, and P.R. Sala, CERN-2005-10 (2005), INFN/TC_05/11, SLAC-R-773.
4. http://www.fluka.org/

Pebble bed reactor core modelling

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Pebble bed type high-temperature reactors could be a viable option to provide high-temperature heat for industrial processes in addition to electricity if the economic feasibility of the concept can be proved. A well-designed pebble bed reactor also has unique safety features that do not require active safety systems. However, some open safety-related questions remain, such as, the possibility for massive oxidation of graphite fuel elements due to air-ingress into the core and the possible transport of radioactivity with graphite dust that forms in the core during operation.
In order to evaluate and prove the safety of any reactor design, well validated and reliable computer codes are needed. Generation IV reactors introduce entirely new challenges to modelling as they use exotic coolants and fuel designs. Such is the case especially with the pebble bed reactor where the fuel is in the form of coated particles inside spherical graphite elements that are re-circulated through the core. New modelling approaches are needed to model the reactor physics and thermal-hydraulics of such a reactor.
At Lappeenranta University of Technology work is being done to develop methods to model the fuel pebble interactions, reactor physics and thermal-hydraulics. In order to model the packing of fuel pebbles inside the reactor core, discrete element method is used. The method applies specific force models in resolving the interactions between individual pebbles as they are in contact with each other. Simulation propagates in constant time steps and the trajectory of a pebble is calculated after the net forces acting on it are evaluated. There is an in-house code originally developed for the modelling of granular materials in general, that has been modified and adopted for the modelling of pebble bed reactors. The code has been used for packing simulations and most recently for an earthquake transient simulation. For detailed reactor physics modelling, the Monte Carlo reactor physics code Serpent is used. The fuel elements and even the individual coated fuel particles can be defined in detail with Serpent. Serpent has so far been successfully used for full-core calculations involving 50,000 stochastically distributed fuel pebbles, which are read in from the output of a discrete element method simulation. As the first validation case for the reactor physics method, the Russian ASTRA critical pebble bed experiments have been calculated. The full-core thermal-hydraulic model uses a porous medium approach, where a coarse calculation grid is formed inside the flow domain. The fuel pebbles form a volume blockage to the flow that is taken into account as a sink term in the momentum equation. For heat transfer, a simple thermal-equilibrium model is currently used but a more realistic non-equilibrium model will be developed in the future work. Correlations available in literature have been used for defining the porosity distribution in the flow domain so far, but work is underway to use the results of discrete element method calculations to define an accurate three dimensional porosity profile. The thermal-hydraulic model has been developed on top of the ANSYS FLUENT general-purpose computational fluid dynamics software with user-defined functions.
The most recent efforts include the development of a coupling between the reactor physics and thermal-hydraulics. A coupling code has been written with Perl that handles the data transfer between Serpent and Fluent and determines the solution convergence. The main objective is to develop a one-of-its-kind code system coupling the most advanced modelling methods of each individual physical process.

6. GEN IV ACTIVITIES IN THE PARTNER ORGANISATIONS

The Gen-IV activities in the Nordic countries have previously been described by Clara Anghel and Sami Penttilä in the NKS report no. 216 (accessible on the NKS website). Here, we will give an update on the partner organizations only. Since NRG, JRC, CVR will be partners from 2012, they are already included here.

6.1 IFE (Institutt for Energiteknikk)

The Halden Project is a joint undertaking of national organizations in 18 countries sponsoring a jointly financed programme under the auspices of the OECD - Nuclear Energy Agency.

The programmes are to generate key information for safety and licensing assessments and aim at providing:

- Extended fuel utilization: Basic data on how the fuel performs in commercial reactors, both at normal operation and transient conditions, with emphasis on extended fuel utilization.
- Degradation of core materials: Knowledge of plant materials behaviour under the combined deteriorating effects of water chemistry and nuclear environment.
- Man-Machine Systems: Advances in computerized surveillance systems, human factors and man-machine interaction in support of upgraded control rooms. These are collectively known as The Joint Programme.

Recently, Gen IV related research has been taken up as part of the Joint Program. IFE/Halden will continue its development of Gen-IV relevant instruments and plasma coatings for corrosion protection. Further, IFE-Halden will continue its efforts to find funding for the construction of an in-pile supercritical water loop in the Halden reactor. In addition, in-pile irradiations of instruments and materials for Gen-IV will be performed in the Halden reactor. Though IFE-Halden’s focus is on supercritical water, instruments for use in liquid sodium and lead are also being developed. Prototype instruments for use up to 700 C will be built in the course of 2012. Further, Halden follows closely (as member of the user group) the research going on within the European Program GETMAT (GenIV and Transmutation of Materials).

6.2 VTT (Technical Research Centre of Finland)

At the moment, VTT's Gen IV materials research is focusing mainly on SCWR concept study. Part of the Gen IV development is concept specific but many are still common to all or most of the Gen- IV concepts. Thus, materials cross-cutting aspects have arisen in many projects as the main theme. Especially increased interest towards convergence of the Gen IV material research in Europe has launched material specific cooperation in EERA (European Energy Research Alliance) where VTT is a member. The EERA group has been established in 2009 regrouping several scientific representatives from research institutions around Europe with the purpose to foster and perform material studies for nuclear applications. In addition, VTT is closely connected with Gen-IV field EU projects such as GETMAT, SCWR-FQT (Supercritical Water Reactor Fuel Qualification Test) and MATTER (MATerial TEsting and Rules) as well as the Jules Horowitz Reactor and Academy of Finland project NETNUC (New Type Nuclear Reactors project in Sustainable Energy (SusEn) research programme of Academy of Finland).

6.3 Studsvik Nuclear

Studsvik Nuclear´s activities are currently focused on optimizing the operation efficiency and safety of existing nuclear reactors. This includes for example;

- Materials testing for qualification of newly developed fuel claddings
- Mechanical testing and microanalysis of irradiated material to study performance during operation.
- Failure analysis from failed parts of the nuclear reactor such as fuel cladding, control rods, spacers, steam turbines, reactor tank lid etc
- Surveillance tests to predict the lifetime of the reactor vessel.
- Studies under anticipated accidents like LOCA (loss of coolant) and RIA (Reactivity Induced Accidents).
- Interim storage and final disposal studies
- Management of large international programs including programs performed in cooperation with OECD/NEA, IAEA and NRC

With 60 years of experience of nuclear technology and radiological studies, Studsvik can contribute with knowledge transfer and “lessons learned” from the current generation of nuclear reactors to Generation IV. Parts from numerous Nuclear reactors all over the world have been investigated in Studsvik over the years and the root cause of many failures has been identified. Studsvik therefore has extensive knowledge of how irradiation affects materials behavior as well as how to overcome difficulties encountered when testing irradiated materials in hot cells. These are experiences that can be directly applied to Gen IV. Knowledge transfer can be made through seminars, workshops and consultation when performing mechanical tests, in-pile tests and microanalysis or when qualifying fuel codes.

6.4 Risø, DTU

Risø DTU’s activities within reactor physics/nuclear power technologies are presently directed towards neutronics simulations for the European Spallation Source (ESS), to be built in Lund, Sweden. The Materials Research Division (AFM) of Risø DTU has strong competences in neutron scattering and imaging for materials characterization. Development of cutting edge experimental techniques as well as optimization and maintenance of the scattering simulation software McStas are focal areas in the Division. For ESS, AFM plans to contribute by spectrometer design and implementation as well as by supply of McStas and relevant data analysis software and will in due time become users of the facility.

Besides these activities AFM have competences and interests in materials development and in understanding and optimizing mechanical and thermal reactions in metals and alloys; examples are nanometals and ODS alloys which are also studied for possible use in fusion reactors. Risø DTU has collaborated with the Halden Reactor Project in the conceptual design of a SCWR loop, and plans to continue this activity. The collaboration aims at the active participation of PhD or MSc students and will benefit training of young researchers in the field of nuclear energy.

6.5 NRG (Nuclear Services for Energy, Environment & Health)

The Nuclear Research and consultancy group (NRG) is the center of nuclear expertise in the Netherlands. NRG develops knowledge, processes and products for safe application of nuclear technology in the areas of energy, environment, and health. This includes technology development for safe and economic operation of future nuclear energy systems, such as GenIV reactors. NRG performs a range of fuel and material irradiations at the High Flux Reactor in Petten.

NRG is currently participating in the following EU projects: GETMAT (Gen IV and Transmutation Materials) and MATTER (MATerial TEsting and Rules), and ARCHER (HTR). Within the GETMAT project NRG has gained a large experience in performing irradiation experiments in liquid metal (Lead Bismuth Eutectic (LBE)) and in post irradiation materials characterization including fracture toughness testing and electron microscopy. NRG is also involved in the development of new non-melting joining techniques for high temperature materials such as Oxide Dispersion Strengthened (ODS) steels. In MATTER NRG will develop a fracture toughness testing procedure in conditioned lead-based liquid metal and perform fracture toughness testing of 9Cr steels in LBE. The collaboration aims at the future development of joint activities.

6.6 JRC IET (Joint Research Centre – Institute for Energy and Transport)

JRC IET Petten covers the GEN-IV material research activities via action called MATTINO (MATerials performance assessmenT for safety and Innovative Nuclear reactors)

MATTINO covers the following range of multi-annual R&D topics to support long-term EU policy needs and to ensure JRC competences in nuclear safety technology for all reactor types of relevance for Europe:

1. Advanced thermo-mechanical, corrosion resistance, and irradiation and environmental safety performance assessment of structural materials, incl. joints/welds and coated systems, taking into account high temperature coolant compatibility and long-term operation;

2. Development of codes-of-practice for advanced non-standard thermo-mechanical, miniaturised specimen testing, and environmental testing; harmonisation of test methods, inspection procedures and data management tools applied in Europe; participation in round robins and input to standardisation bodies;

3. Structural materials performance assessment under anticipated operational occurrences and accidental conditions for safety issues; input to material design codes and standards;

4. Physically-based modelling with experimental validation to contribute to a better understanding of the materials performance in the respective conditions and environments, thus reducing the need for large-scale and expensive experiments;

5. Basic assessment of different reactors nuclear systems needed for an overall safety assessment and the role of nuclear energy in future energy systems;

6. International cooperation through EERA, NULIFE, SNETP, ESNII, GIF, IAEA, I-NERI, NeT, IAEC as well as through the participation in various competitive actions.

The specific objectives have been chosen to strike a proper balance between experimental research into materials performance assessment and the need for JRC to provide a European reference function through the execution of pre-normative R&D and the participation in Materials Codes and Standards development. Furthermore, the need for materials performance modelling and simulation is taken into account by a dedicated horizontal topic.

The Deliverables of the Annual Work Programme 2012 are defined under five pillars:
1. Safety assessment of components and systems for design and operations;
2. pre-normative research for innovation;
3. modelling and simulation;
4. system integration;
5. international collaboration.

The first pillar is overarching and exploits the results of the other pillars. It addresses primarily the results with respect to safety rather than the methodology.

Standardisation is a vehicle to innovation. MATTINO is addressing pre-normative research into the harmonisation and standardisation of innovative materials testing techniques and data management among European research institutions as well as for the development of design codes.

The development of robust and predictive physical models and simulation tools can accelerate the materials development and qualification processes, and is necessary for extrapolating outside known domains of operational experience. Modelling and simulation and experimental work are complementary and need to be fully integrated. Three main areas for model development have been singled out: (1) high-temperature applications with creep and fatigue, (2) integrity of cladding tubes and (3) stress corrosion cracking.

Specific activities, such as stress corrosion cracking, support several of the five main objectives and appear as sub-objectives. A large effort is dedicated to oxide dispersion strengthened (ODS) steels with excellent high temperature strength and corrosion resistance and, hence, a high potential for nuclear applications in innovative concepts. In WP2012 MATTINO will investigate new charges of ODS steels within various international collaboration agreements.

International research efforts are required for harmonized procedures for plant life management of today's reactors and to qualify materials and components for innovative reactors. MATTINO activities form part of the Euratom contribution to GIF, and are linked to the SNETP Strategic Research Agenda (SRA), NULIFE and ESNII. Close international collaboration also exists through various RTD funded competitive actions and IAEA Coordinated Research Projects. Most importantly, WP 2012 will contribute to the Joint Programme on Structural Materials for Innovative Nuclear Systems of the European Energy Research Alliance (EERA) in support of the SET-Plan targets and to the activities for present reactors in NULIFE. Finally MATTINO will support the key policy directorates with independent assessment of reactor safety issues related to materials and components.

In 2012 MATTINO will contribute to the following FP7 projects: GETMAT, STYLE, ISP1, ARCHER, MATTER, SCWR-FQT, MULTIMETAL, LONGLIFE, FAIRFUELS, PELGRIMM.

6.7 CVR (Nuclear Research Centre Rez)

The present activities carried out in CVR (Research Centre Rez) in the field of Gen IV are focused mainly on the development, construction and operation of experimental in-pile loops for operation in its LVR-15 material testing reactor. Two facilities have already been built and installed out-of-pile for trial operation; those are the Supercritical Water Loop (SCWL) and the High Temperature Helium Loop (HTHL). For some of the concepts with fast neutron spectra, such as the GFR and SFR, CVR carries out theoretical simulations and calculations in support of development of conversion cycles with supercritical CO2. For all the above mentioned scopes, CVR participates in Gen-IV related EU projects, such as SCWR-FQT (Supercritical Water Reactor –Fuel Qualification Test), where CVR acts as the project coordinator, MATTER (MATerial TEsting and Rules), ARCHER (Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D), EURECA! (Cooperation between EU and Canada in Education, Training and Knowledge Management on Super-Critical Water Reactors), and has also applied for a large project called SUStainable ENergy (SUSEN), where additional infrastructure in support of the Gen-IV systems is foreseen to be built. A collaboration of mutual benefit is currently foreseen between CVR and IFE in which electrochemical sensors developed in IFE would be tested in-pile in the LVR-15 reactor.

In-pile material research in CVR is based on its past experience in performing experiments in PWR and BWR simulating loops, five of which have been operated in the LVR-15 reactor in the past.

7. PLANNED ACTIVITIES FOR 2012

Activities planned for NORDIC-GEN4 during 2012 are:

- A seminar for Generation IV Nuclear Energy systems will be held at Risø DTU
- Initiate collaborative projects
- Set up a new website

8. ACKNOWLEDGEMENTS

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9. REFERENCES

1. T. Abram, S. Ion, Generation-IV nuclear power: A review of the state of the science, Energy Policy 36 (2008) 4323-4330.

2. K.L. Murty, I. Charit, Structural materials for Gen-IV nuclear reactors: Challenges and opportunities, Journal of Nuclear Materials 383 (2008) 189-195.

3. C. Fazio, et. al., European cross-cutting research on structural materials for Generation IV and transmutation systems, Journal of Nuclear Materials 392 (2009) 316-323.

4. J.G. Marques, Evolution of nuclear fission reactors: Third generation and beyond, Energy Conversion and Management 51 (2010) 1774-1780.

5. B. Raj, Materials and Manufacturing Technologies for Sodium Cooled Fast Reactors and Associated Fuel Cycle: Innovations and Maturity

6. C.R.F. Azevedo, Selection of fuel cladding materials for nuclear fission reactors, Engineering Failure Analysis 18 (2011) 1943-1962

7. G.L. Fiorini, A. Vasile, European Commission – 7th Framework Programme, The Collaborative Project on European Fast Reactor (CP ESFR), Nuclear Engineering and Design 241 (2011) 3461-3469

8. L. Cinotti, et. al., Lead-cooled system design and challenges in the frame of Generation IV International Forum, Journal of Nuclear Materials 415 (2011) 245-253

APPENDIX A: AGENDA OF THE NOMAGE4 SEMINAR IN HALDEN (2011)

30 October

18:30-21:00: Reception at Park hotel

31 October

8:15	Registration	
	Chairman: Rudi Van Nieuwenhove	
8:30	*Introduction to IFE and the Halden Reactor Project* *The Halden Project aims to increase the collaboration with Universities*	Margaret McGrath, IFE Fridtjov Øwre, IFE
8:55	*European Program on Materials Research for Generation IV reactors*	Fazio Concetta, KIT
9:20	*The NOMAGE4 network*	Rudi Van Nieuwenhove, IFE
9:45	*The Swedish industry's role and view on GenIV*	Annika Olsson, Åma Uppsala University
10:10	*The Swedish GENIUS project*	Peter Szakalos, KTH, Ane Håkansson, Uppsala University, Christian Ekberg, Chalmers University of Technology
10:35	Coffee break	
	Chairman: Serguei Gavrilov	
11:00	*MYRRHA: An Innovative and Unique Research Facility*	Hamid Ait Abderrahim, SCK-CEN
11:25	*Gen-IV research in Finland*	Riita Kyrki-Rajamäki, Lappeenranta University of Technology, LUT Energy
11:50	*Transient Analysis of the European Lead Cooled System ELSY*	Milan Tesinsky, KTH
12:15	**Lunch**	
	Chairman: Serguei Gavrilov	
13:30	*Transmutation of Americium in Sodium Cooled Fast Reactors*	Youpeng Zhang, KTH
13:55	*Materials for nuclear applications*	Sten Anders Wilson, Sandvik
14:20	*Canadian R&D Effort on Gen IV materials*	Wenyue Zheng, NRCan, CANMET-MTL
14:45	**Posters**	
15:10	End presentations	
15:30	Visit to reactor site	
17:15	Visit to workshop	
19:00	Dinner at Haldens Klub	

1st November

	Chairman: Peter Szakalos	
8:15	*Evaluation of corrosion resistance of experimental FeCrAl alloys in liquid Pb*	Jesper Ejenstam, KTH
8:40	*Material issues for design and licensing of MYRRHA ADS system*	Serguei Gavrilov, SCK-CEN
9:05	*Activities of the HRP into research for Generation IV reactors*	Rudi Van Nieuwenhove, IFE
9:30	*In-pile test of a small scale fuel assembly under supercritical water conditions*	Thomas Schulenberg, KIT
9:55	*Effect of surface modification on the corrosion resistance of 316L in SCW*	Sami Penttilä, VTT
10:20	*Water chemistry for SCWR*	Markéta Zychova, Research Centre Rez Ltd
10:45	Coffee break	
	Chairman: Teodora Retegan	
11:00	*Corrosion and SCC material testing in SCW in JRC IE Petten*	Radek Novotny, JRC IE Petten
11:25	*Spent nuclear fuel recycling using PRISM*	David Powell, GE Hitachi
11:50	*Advanced Radiation Resistance Materials (ARRM) Development Program for LWR Applications* *replaced by: Pulsed Plasma Surface Treatment*	Larry Nelson, JLN Consulting Rune Hoel
12:15	*The potential use of thorium-based fuels in existing reactor designs.*	Sunniva Rose , Sunniva Siem Univ. of Oslo, Jon Wilson, IPN Orsay
12:40	**Lunch**	
	Chairman: Sami Penttilä	
13:40	*Testing of Fuels and Materials for Next Generation Reactors at HFR Petten*	Jaap van der Laan, NRG, Petten
14:05	*Extraction of Th from Norwegian mineral resources*	Tor Bjørnstad, IFE
14:30	*In-pile and out-of pile testing to predict fuel cladding failures*	Anna-Maria Alvarez ,Studsvik Nuclear
14:55	*Material characterization capabilities at IFE-Kjeller*	Barbara Oberländer, IFE
15:20	End	

Posters:

Øivind Berg and Svein Nøvik, IFE/HRP, *Thorium resources in Norway and potential application in Generation-IV reactors*
Risto Vanhanen, Aalto University, *Assessment of Feasibility of Thorium Fuel in BWRs*
Rune Hoel, MOTecH Plasma a.s, *Pulsed plasma surface treatment*
Aarnio Pertti, *Simulating transmutation in Myrrha with Fluka MC-code*
Heikki Suikkanen, Ville Rintala, *Pebble bed reactor core modeling*

Title	NOMAGE4 activities 2011, Part I, Nordic Nuclear Materials Forum for Generation IV Reactors: Status and activities in 2011
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Abstract

A network for materials issues has been initiated in 2009 within the Nordic countries. The original objectives of the Generation IV Nordic Nuclear Materials Forum (NOMAGE4) were to form the basis of a sustainable forum for Gen-IV issues, especially focusing on fuels, cladding, structural materials and coolant interaction. Over the last years, other issues such as reactor physics, thermal hydraulics, safety and waste have gained in importance (within the network) and therefore the scope of the forum has been enlarged and a more appropriate and more general name, NORDIC-GEN4, has been chosen for the forum. Further, the interaction with non-Nordic countries (such as The Netherlands (JRC, NRG) and Czech Republic (CVR)) will be increased.

Within the NOMAGE4 project, a seminar was organized by IFE-Halden during 30 November – 1 November 2011. The seminar attracted 65 participants from 12 countries. The seminar provided a forum for exchange of information, discussion on future research reactor needs and networking of experts on Generation IV reactor concepts. The participants could also visit the Halden reactor site and the workshop.

Key words NOMAGE4, Nordic-Gen4, seminar, Halden

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