SAFIR2018 INTERIM SEMINAR AND WEBINAR 23.-24.3.2017

SAFIR2018 - The Finnish Research Programme on Nuclear Power Plant Safety 2015 - 2018

Day 1: 23.3.2017

Chair: Jorma Aurela, MEAE

8:30 – 9:00	**Registration**
9:00 – 9:40	**Opening session: Liisa Heikinheimo (MEAE), Petteri Tiippana (STUK), Jari Hämäläinen (SAFIR2018)**
9:40 – 10:30	**Research area "RG1 Automation, Organisation and Human Factors"**
9:40 – 10:00	*Crafting operational resilience in nuclear domain (VTT, FIOH),* Jari Laarni
10:00 – 10:20	*Integrated safety assessment and justification of nuclear power plant automation (VTT, Aalto, FISMA, Risk Pilot, IntoWorks),* Antti Pakonen
10:20 – 10:30	Poster presentation: *Management principles and safety culture in complex projects (VTT, Aalto, University of Oulu),* Nadezhda Gotcheva
10:30 – 11:00	**Coffee break**
11:00 – 12:20	**Research area "RG2 Severe Accidents and Risk Analysis"**
11:00 – 11:20	*Experimental and numerical methods for external event assessment improving safety (VTT), Alexis Fedoroff*
11:20 – 11:40	*Comprehensive analysis of severe accidents (VTT),* Anna Nieminen
11:40 – 12:00	*Chemistry and transport of fission products (VTT),* Teemu Kärkelä
12:00 – 12:20	*Probabilistic risk assessment method development and applications (VTT, Aalto, Risk Pilot),* Ilkka Karanta
12:20 – 13:20	**Lunch**

Chair: Sami Hautakangas, Fortum

13:20 – 13:50	**Research area "RG2 Severe Accidents and Risk Analysis" (cont.)**
13:20 – 13:30	Poster presentation: *Extreme weather and nuclear power plants (FMI),* Kirsti Jylhä
13:30 – 13:40	Poster presentation: *Fire risk evaluation and defence-in-depth (VTT, Aalto),* Anna Matala
13:40 – 13:50	Poster presentation: *Safety of new reactor technologies (VTT),* Jarno Kolehmainen
13:50 – 14:10	**Research area "RG3 Reactor and Fuel"**
13:50 – 14:00	Poster presentation: *Nuclear criticality and safety analyses preparedness at VTT (VTT),* Pauli Juutilainen
14:00 – 14:10	Poster presentation: *Neutronics, burnup and nuclear fuel (Aalto),* Jarmo Ala-Heikkilä
14:10 – 15:10	**Poster session and coffee break** (poster presenters of research areas RG1, RG2 and RG3 shall be available for discussions)
15:10 – 16:10	**Research area "RG3 Reactor and Fuel"**
15:10 – 15:30	*Development of a Monte Carlo based calculation sequence for reactor core safety analyses (VTT*), Jaakko Leppänen
15:30 – 15:50	*Physics and chemistry of nuclear fuel (VTT),* Ville Tulkki
15:50 – 16:10	*Safety analyses for dynamical events (VTT),* Ville Sahlberg
16:10 – 16:20	**Break**
16:20 – 17:10	**Invited presentations**
16:20 – 16:40	Impacts of Fukushima Daiichi accident on Nuclear Safety, Petteri Tiippana (STUK)
16:40 – 17:00	Implications of Fukushima Daiichi accident on Nuclear Safety Research, Ilona Lindholm (VTT)
17:00 – 17:10	Discussion and break

Chair: Jorma Aurela, MEAE

17:10 – 18:20	**Feedback on SAFIR2018 – Panel discussion**
17:10 – 17:20	Radiation and Nuclear Safety Authority (STUK)
17:20 – 17:30	Fennovoima
17:30 – 17:40	Fortum
17:40 – 17:50	TVO
17:50 – 18:20	Discussion
18:30 –	**Buffet dinner**

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Day 2: 24.3.2017

Chair: Antti Tarkiainen, TVO

8:30	**Opening**
8:30 – 9:50	**Research area "RG4 Thermal hydraulics"**
8:30 – 8:50	*Couplings and instabilities in reactor systems (LUT),* Markku Puustinen
8:50 – 9:10	*Integral and separate effects tests on thermal-hydraulic problems in reactors (LUT),* Virpi Kouhia
9:10 – 9:30	*Development and Validation of CFD Methods for Nuclear Reactor Safety Assessment (VTT, Aalto, LUT),* Timo Pättikangas
9:30 – 9:40	Poster presentation: *Comprehensive and systematic validation of independent safety analysis tools (VTT),* Seppo Hillberg
9:40 – 9:50	Poster presentation: *Uncertainty and sensitivity analyses for reactor safety (VTT, Aalto),* Ville Valtavirta
9:50 – 10:20	**Invited presentation**
9:50 – 10:10	Nordic Co-Operation in Nuclear Safety Research, Monika Adsten (Energiforsk)
10:10 – 10:20	Discussion
10:20 – 10:50	**Coffee break**
10:50 – 12:10	**Research area "RG5 Structural Integrity"**
10:50 – 11:10	*Analysis of fatigue and other cumulative ageing to extend lifetime (VTT, Aalto),* Otso Cronvall
11:10 – 11:30	*Long term operation aspects of structural integrity (VTT),* Sebastian Lindqvist
11:30 – 11:50	*Mitigation of cracking through advanced water chemistry (VTT),* Konsta Sipilä
11:50 – 12:10	*Effect of thermal ageing on the long-term behaviour of Alloy 690 (VTT, Aalto),* Roman Mouginot
12:10 – 13:10	**Lunch**

Chair: Tomi Routamo, STUK

13:10 – 13:30	**Research area "RG5 Structural Integrity" (cont.)**
13:10 – 13:20	Poster presentation: *Condition monitoring, thermal and radiation degradation of polymers inside NPP containments (VTT, SP),* Konsta Sipilä
13:20 – 13:30	Poster presentation: *Non-destructive examination of NPP primary circuit components and concrete infrastructure (VTT, Aalto),* Tuomas Koskinen
13:30 – 14:00	**Research area "RG6 Research Infrastructure"**
13:30 – 14:00	*Jules Horowitz Reactor - JHR collaboration & Melodie follow-up (VTT),* Santtu Huotilainen
14:00 – 15:00	**Poster session and coffee break** (poster presenters of research areas RG4 and RG5 shall be available for discussions)
15:00 – 16:00	**Research area "RG6 Research Infrastructure" (cont.)**
15:00 – 15:20	*Development of thermal-hydraulic infrastructure at LUT (LUT),* Joonas Telkkä
15:20 – 16:00	*Radiological laboratory commissioning 2016 (VTT),* Wade Karlsen
16:00 – 16:30	**Final discussion**
16:00 – 16:30	Final discussion moderated by the chairperson of SAFIR2018 Management Board Marja-Leena Järvinen
16:30	**Closing**

Poster sessions:

Research area "RG1 Automation, Organisation and Human Factors"
Management principles and safety culture in complex projects (VTT, Aalto, University of Oulu), Nadezhda Gotcheva

Research area "RG2 Severe Accidents and Risk Analysis"
Extreme weather and nuclear power plants (FMI), Kirsti Jylhä
Fire risk evaluation and defence-in-depth (VTT, Aalto), Anna Matala
Safety of new reactor technologies (VTT), Jarno Kolehmainen

Research area "RG3 Reactor and Fuel"
Nuclear criticality and safety analyses preparedness at VTT (VTT), Pauli Juutilainen
Neutronics, burnup and nuclear fuel (Aalto), Jarmo Ala-Heikkilä

Research area "RG4 Thermal hydraulics"
Comprehensive and systematic validation of independent safety analysis tools (VTT), Seppo Hillberg
Uncertainty and sensitivity analyses for reactor safety (VTT, Aalto), Ville Valtavirta

Research area "RG5 Structural Integrity"
Condition monitoring, thermal and radiation degradation of polymers inside NPP containments (VTT, SP), Konsta Sipilä
Non-destructive examination of NPP primary circuit components and concrete infrastructure (VTT, Aalto), Tuomas Koskinen

VTT TVO Fortum FENNO VOIMA STUK
Ministry of Economic Affairs and Employment of Finland
LUT Lappeenranta University of Technology
Aalto University
Tekes
Strål säkerhets myndigheten
Swedish Radiation Safety Authority

Project Abstracts

Research area "RG1 Automation, Organisation and Human Factors":

CORE - Crafting operational resilience in nuclear domain
The CORE project aims to identify the key characteristics of operational resilience at three defence levels of prevention, preparation, and consequence management and to study how these characteristics are manifested in operator behaviour and how they can be operationalized and measured. The project is divided into six work packages with the aim to consider resilience from various perspectives. We have developed 1) tools and practices for gathering positive operating experiences from challenging operational situations; 2) training interventions and guidance to promote reflective thinking and learning and effective interruption management and troubleshooting among operators; 3) interventions and guidance for the management of acute stress and fatigue; and 4) guidance for the promotion of communication and coordination of activities in emergency exercises. In addition, a safety management procedure, Human Factors tool, has been developed to analyse human contribution to nuclear safety.

SAUNA - Integrated safety assessment and justification of nuclear power plant automation
The general objective of SAUNA (2015-2018) is to develop an integrated framework for safety assessment and transparent safety demonstration of nuclear power plant instrumentation and control (I&C) systems. In the first two project years, SAUNA has specified reference models for Systems Engineering life-cycle processes, with focus on I&C qualification. We have also developed concepts and models for analysing I&C architectures and Defence-in-Depth. In terms of analysis methods and tools, SAUNA has created ways of integrating a formal verification method called model checking with requirement engineering, plant simulation, testing, and probabilistic assessment. We have also developed methods for assessing engineering processes, and studied an interdisciplinary hazard analysis method called STPA. On the topic of structured safety demonstration, a Nordic expert network has been established for collecting and reporting licensing practices, and developing methods for justifying overall safety of nuclear I&C in a transparent way.

MAPS - Management principles and safety culture in complex projects
MAPS aims at enhancing nuclear safety by supporting high quality execution of complex nuclear industry projects. The conceptual governance model for inter-organizational networks we developed provides insight in practical relevance of project governance approaches for enhancing safety. The ongoing case studies identify the need to pay more attention to non-technical aspects of complexity and understand the implications from unexpected events and changes in projects. Benchmarking the Norwegian oil & gas industry proved useful for sharing best practices and challenges on management of complex projects. The co-existence of different subcultures in projects calls for recognizing and aligning emerging tensions to manage cultural complexity. Identifying practical methods to assure and improve safety culture culminated in a preliminary framework for evaluating their applicability in temporary environment. System dynamics modelling represented reinforcing loops that can explain issues, such as delays, in complex nuclear industry projects.

Research area "RG2 Severe Accidents and Risk Analysis":

ERNEST - Experimental and numerical methods for external event assessment improving safety
ERNEST project concentrates mainly on developing, validating and taking into use more reliable methods for external event assessment improving nuclear safety. The work carried out so far has included medium scale impact tests on reinforced concrete slabs for computational model validation, numerical simulations of the previously executed test, numerical simulation of a wide body passenger aircraft impact against a rigid target for impact loading model validation, extension and complementation of existing concrete material models for numerical simulations with Abaqus finite element software, numerical study of a 1/3-scale reactor containment building mock-up under pre-stressing and pressurization and comparison of the calculated results against the ones measured in the tests with the actual mock-up and model development for earthquake induced vibrations starting from the rupture of the fault.

CASA - Comprehensive analysis of severe accidents
Overall understanding of the progress and mitigation of severe accidents was strengthened by simulating the Fukushima accidents for all three units with the integral code MELCOR. A good correspondence was achieved to the available measurement data. Cooling the core melt is the primary objective in all of its locations. Multi-dimensional flooding increases the dryout heat flux of a debris bed but the benefit is lost if the conical bed is more than 1.5 times higher than the cylindrical bed. However, the coolability of a conical bed could be estimated less conservatively by establishing a temperature-based coolability criterion. In the case of an ex-vessel melt pool, the correlations were found to overestimate heat transfer coefficients and that heat flux was proven to be more biased upwards than expected. The new water ingression model in MELCOR produced good results when gas is bubbling through the melt pool. All these findings increase the understanding of core melt management. The biggest threats for the containment integrity in addition to core melt are highly energetic events of steam and hydrogen explosions. The melt drop size, that depends on physical properties of the melt, was found to have the strongest effect on the steam explosion strength. Hydrogen explosions in the Nordic BWR containments were proven to be very unlikely. Well-founded dose estimates are needed for example when licensing the operation of instrumentation and automation systems. Dose rates calculated with the ASTEC and NRC methods were compared. In all cases the total dose rate estimates are within a factor of two that can be considered rather acceptable. If the containment integrity is lost, it is necessary to assess the transport of radioactive release to evaluate environmental consequences. It is important to include ingestion dose to offsite dose assessment because its contribution dominates the total dose. Calculations indicate that if the release magnitude corresponds to the national severe accident release limit (100 TBq of Cs-137), countermeasures are improbable beyond the distance of 20 km.

CATFIS - Chemistry and transport of fission products
The aim in the CATFIS project during 2015-2016 was to investigate the transport and chemistry of gaseous and particulate fission products in severe accident conditions. The main focus was on the behaviour of iodine and ruthenium which are highly radiotoxic and the mitigation of their possible source term is of utmost importance. It was observed that the fission product deposits on the reactor coolant system surfaces act as an important source of gaseous iodine, which can enhance the iodine source term. In air ingress conditions, the oxidizing air radiolysis products had a significant impact on the formation of gaseous ruthenium, which was observed to exist even at the low temperatures of containment building. Similarly, airborne CsI reacted with ruthenium oxides and increased the gaseous ruthenium fraction. The follow-up of OECD/NEA STEM-2 and BIP-3 projects was carried out.

PRAMEA - Probabilistic risk assessment method development and applications
PRAMEA aims to generate new knowledge in probabilistic risk assessment (PRA), covering most research related to it in the SAFIR2018 program. In human reliability assessment (HRA), research has concentrated mostly on the development of a framework for conducting HRA in the context of advanced (digitalized) control rooms in nuclear power plants. Research on multi-unit PRA has produced information on state of the art in the field, and a proposed framework for carrying out multi-unit PRA analyses. Research on dynamic PRA has produced information about the applications of dynamic flowgraph methodology, a method for conducting dynamic PRA modelling and analyses. Research on level 2 PRA has created a summary of VTT's recent research on the topic, and assessment of factors affecting release height and temperature in severe accidents of nuclear power plants, together with methods for estimating these. Research on level 2 PRA has been carried out on two fronts. Methods and implementation of integrating risks, conducting statistical analyses in level 2, and integrating level 1 and level 2 analyses to provide a richer picture of the effect of factors from level 1 to level 2 results, have been developed. In the PRAMEA part of a Nordic cooperation project, methods for conducting level 3 analyses by integrating level 2 and level 3 methods, and probabilistic and deterministic analyses have been studied and developed, producing a pilot study on the Fukushima accident. Also a guideline document for conducting level 3 analyses has been produced in the Nordic project. Recently used methods for population dose assessment have been reviewed.

EXWE - Extreme weather and nuclear power plants
Estimates of probabilities of hazards related to external events are needed for overall safety management of nuclear power plants (NPPs). Historical time series of extreme convective weather events showed notable cross correlations between summertime 1) thunderstorm days, lightning and heavy rainfall, 2) heavy rainfall and tornadoes and 3) large hail and tornadoes. Two of four classes of synoptic-scale circulation patterns that favour significant hail (5 cm or larger) have similarities with the patterns observed in Finland in tornado situations. The record-breaking sea-effect snowfall case at Merikarvia on 8 January 2016 was well captured by the HARMONIE weather prediction model, compared to weather radar images. Even if the windstorm climate in Finland remains the same, the impacts of the storms will alter due to the ongoing change in environmental conditions. Recent studies on global sea level rise have not revealed new information that would imply major changes on the mean sea level scenarios for the Finnish coast. The local flooding risk estimates can be improved by investigating the maximum elevation of the continuous water mass on the shore by combining the effects of sea level and wind-generated waves. Rapid sea level oscillations are caused by different meteorological conditions during summer and winter seasons. Although no serious effects have occurred in Finland, large geomagnetically induced currents must be taken into account when installing new transformers or power lines affecting the operation of NPPs. High-resolution meteorological modelling has progressed to a stage where a proper assessment of coastal meteorological conditions can be fed into a dispersion modelling system.

FIRED - Fire risk evaluation and defence-in-depth
A significant proportion of the overall core damage risk in nuclear power plants (NPP) is associated with internal fires. In addition, a fire on NPP can cause large financial losses even if the risk to the reactor safety was small. Therefore the possible initiating event scenarios and the operation of defence-in-depth after ignition are important topics in the research of nuclear safety. The computational tools that are used for assessing the fire risks have developed significantly over the last ten years: The deterministic analyses are increasingly based on CFD and the probabilistic analyses using Monte Carlo simulation have been carried out. These developments have improved the reliability and accuracy of the safety analyses. In FIRED project, the research on fire risks and defence-in-depth is focused on three main topics: evaluating the fire risks of cables during plant life cycle, assessing the performance of fire-barriers, and development, maintenance and validation of fire simulation tools.

GENXFIN - Safety of new reactor technologies
The main mission of the GENXFIN project is to improve scientific and technologic expertise in the field of new nuclear energy technologies and related processes through international collaboration. The main objective is to coordinate participation in various international working groups and information dissemination on interested parties. Essential part of the project was to get familiar with the licensing of innovative Small Modular Reactor (SMR) concepts which is interesting from a national perspective. Material Research on new reactor technologies has an educational role in Finland but it is also a platform for technology development. In order to mitigate the worst effects of climate change the whole energy sector needs to be decarbonized.

Research area “RG3 Reactor and Fuel”:

KATVE - Nuclear criticality and safety analyses preparedness at VTT
Developing the capabilities in criticality safety, radiation shielding, activation analysis, dosimetry and dry storage modelling has been the main objective in the KATVE project. The development consists of both code and method development as well as educating new experts. The work on criticality safety has mostly consisted of assembling the validation package for fresh fuel criticality analysis and creating competence to utilize burnup credit in the spent fuel criticality safety studies. Photon transport routines have been developed to the Serpent Monte Carlo code in order to expand its applicability from reactor physics and burnup calculation to various radiation shielding applications. Dry storage cask has been a new research subject in the Finnish scope. The heat transfer in such a cask has been computed to determine the peak cladding temperature of the stored fuel pins as a function of time. It was found that the temperature is kept below the given limit of 400 °C three and half years after discharging from the core. The activation analysis and dosimetry capabilities have been maintained: the MAVRIC code has been taken into use and new experts have been educated to use the dosimetry calculation chain.

NEPAL15 - Neutronics, burnup and nuclear fuel
This project was a direct one-year continuation of the NEPAL project of the SAFIR2014 programme. Our main focus area has been accurate burnup calculations that aim at finding rare but potentially problematic nuclides like strong absorbers or other reactor-physically important nuclides. New burnup calculation methods have been developed, implemented mainly in the Serpent code, and thoroughly evaluated. Evaluations show significant improvements in accuracy without additional computational resources. The work has been published in open scientific literature and at conferences. In these projects, we have also developed a novel mesoscopic model of the thermal creep failure of fuel pellets. The model includes damage accumulation from radiation-induced fission gas buildup and the behaviour of the gases themselves. Additionally, optimal reconstruction parameters for cross section libraries of Serpent have been investigated.

MONSOON - Development of a Monte Carlo based calculation sequence for reactor core safety analyses
The MONSOON project continues the development of the Serpent Monte Carlo code, with the specific goal of establishing a complete and independent calculation sequence for the safety analyses of Finnish power reactors. The work consists of developing methodologies for spatial homogenization, which forms the first part of the traditional multi-stage calculation scheme applied for core physics calculations. The work was started already in the KÄÄRME project of SAFIR2014, which provided a proof-of-concept type demonstration for the use of the continuous-energy Monte Carlo method for spatial homogenization. The MONSOON project aims to proceed from feasibility studies to practical applications and routine full-scale safety analyses. Serpent development is carried out in close collaboration with a large international user community. The code is currently used in more than 170 organizations in 37 countries around the world.

PANCHO - Physics and chemistry of nuclear fuel
PANCHO has been the essential platform for the ongoing development of FINIX fuel behaviour module and its accompanying automatic validation system. The studies on fuel behaviour in accident conditions have been framed within the international projects, where the accident simulation codes have been tested and validated. Also a coupling between SCANAIR and GENFLO codes has been developed in the project.

The chemistry experiments aimed to investigate characteristics of initial dissolution of crystalline ThO_2 by adding ^{229}Th tracer into the aqueous phase. The experiments were conducted for total 534 days and are to be complemented by modelling approach. The experimental campaign on cladding creep demonstrates that the VTT equipment is capable of performing suitable transient tests which support the modelling activities. One DSc thesis has been finalized based on the ongoing cladding creep modelling work.

SADE - Safety analyses for dynamical events
In 2015, a pin power reconstruction module was implemented to the transient analysis code TRAB3D and its results were compared to the reactor physics code Serpent 2. The comparison led to the discovery of a need to improve the capabilities to model axial heterogeneities in reactor cores. Master’s thesis on the subject was completed in 2016 and differences between TRAB3D and Serpent 2 predictions were halved. Serpent 2 was used to generate group constants for both TRAB3D and the transient analysis code HEXTRAN.

HEXTRAN and the systems code SMABRE were successfully coupled internally and moved to the Linux computing environment. In addition, the porous-CFD code PORFLO was coupled with HEXTRAN and SMABRE. The fully coupled HEXTRAN-SMABRE-PORFLO simulation framework was established and demonstrated by calculating VVER-440 and VVER-1000 transients.. The results demonstrated the feasibility and advantages of coupling 3D thermal hydraulics with 3D neutronics and system codes.

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Research area “RG4 Thermal hydraulics”:

INSTAB - Couplings and instabilities in reactor systems
In the INSTAB project, mechanisms and efficiency of mixing with different subsystems in the suppression pool have been investigated experimentally in the PPOOLEX facility. Tests with a model of a safety relief valve sparger verified that mixing of a thermally stratified water pool can happen also through an erosion process in addition to internal circulation, if suitable flow conditions prevail. Tests with a model of a residual heat removal system nozzle indicated that orientation of the nozzle plays an important role in the success of the mixing process of a thermally stratified pool. The nozzle injection flow rate, injection water temperature and ΔT in the pool have an effect on the mixing process but that effect is not as dominant as of the nozzle orientation. Preliminary wetwell spray tests in PPOOLEX revealed that mixing of a thermally stratified pool with the help of spray injection from above could be possible. The Effective Momentum Source and Effective Heat Source models, developed and implemented to the GOTHIC code by KTH, have been validated for blowdown pipes, SRV spargers and RHR nozzles largely on the basis of the experiment results obtained in the INSTAB project at LUT. In the CFD simulation exercises of steam injection through a blowdown pipe into a suppression pool, done in the INSTAB project, it has been found out that CFD modelling of a pressurizing two-compartment suppression pool requires that the interfacial area density between the liquid and vapour phases is resolved either by using a very dense computational grid or by applying a special interfacial instability model. In addition, OpenFOAM models for the simulation of direct contact condensation phenomenon have been developed and validated against the INSTAB experiment results at LUT and VTT.

INTEGRA - Integral and separate effects tests on thermal-hydraulic problems in reactors
The objective of the project is to improve the understanding of thermal-hydraulic system behaviour by performing integral and separate effects tests, in particular regarding the impact of non-condensable gases on core cooling /1/ and reliability of natural circulation loop decay heat removal. Carefully designed experiments are the most reliable way to obtain fundamental understanding and reliable data of the phenomena. This data will be used in the development and validation of computer codes for the safety analyses of nuclear power plants. Performing experiments not only requires the hardware and programs controlling the devices and gathering data, but also the knowledge of the system behaviour. Computer analyses with system and CFD codes are needed in the planning of the experiments as well as in post analyses to help understanding the physics in the experiments.

NURESA - Development and Validation of CFD Methods for Nuclear Reactor Safety Assessment
Computational Fluid Dynamics (CFD) methods are developed and validated for the identified most important topics in nuclear reactor safety assessment. International single-phase mixing benchmarks are participated and spray experiments performed at LUT are modelled in co-operation with Swedish partners. Models for the departure from nucleate boiling (DNB) are developed for the OpenFOAM code and co-simulation of NPP components with CFD code and Apros system code are performed.

COVA - Comprehensive and systematic validation of independent safety analysis tools
Apros is a system-scale safety analysis tool developed at VTT in cooperation with Fortum since 1986. Apros is used for safety analyses of light water reactors, and thanks to addition of new advanced features in the recent years, it can also be utilized in analysing generation IV nuclear reactors. As a commercial code Apros has a rigorous and extensive version-validation process whose purpose is to ensure that no unwanted changes or error have been introduced in any of the application areas while introducing the new features or changes of existing features and corrections of detected errors into the new released version.

The overall objective of the COVA project is to improve the state of Apros' validation through a systematic and rigorous approach to the validation process, and also to promote this kind of approach to the validation process. The process enhances the expertise in thermal hydraulic area of Generation II and III LWR reactors and includes as an essential part training of new experts to this relevant area of reactor safety. While the main effort is being carried out using Apros, as it has higher national interest as a self-developed independent and versatile safety analysis tool, U.S. NRC's TRACE is also used in analyses of new experiments and in code-to-code comparisons with Apros.

USVA - Uncertainty and sensitivity analyses for reactor safety
In USVA, multidisciplinary uncertainty and sensitivity studies in reactor safety are done. The most important accomplishments of the project in 2015-2016 include a thorough comparison on statistical sensitivity analysis methods in the context of fuel behavior modelling (Ikonen, 2015, 2016), sensitivity analysis of local uncertainties in a 4-code calculation chain of a large break loss of coolant accident (LB-LOCA) (Arkoma and Ikonen, 2016a,b) and the development of an automated uncertainty analysis calculation system for the CASMO-4 – SIMULATE-3 calculation chain (Pusa, 2016). A Master’s thesis was finalized at Aalto University on the combined uncertainty analysis of coupled neutron transport and fuel behaviour codes (Taavitsainen, 2016).

Research area “RG5 Structural Integrity”:

FOUND - Analysis of fatigue and other cumulative ageing to extend lifetime
A summary of the results obtained in the SAFIR2018 FOUND project during the years 2015-2016 is presented. The project is focused in ageing and failure assessment of NPP components. The main results include a broad review of the ageing of the BWR RPV and its internals, new experimental research on the primary water affected fatigue, reviews and developments of various failure assessment methods, thermal fatigue assessment methods, risk informed methods and methods to assess the dynamic response of piping systems. In addition, novel experimental research and results on the residual stresses in NPP piping components that have been in service for decades are presented.

LOST - Long term operation aspects of structural integrity
The goal of this project is to develop the current structural integrity assessment methods for primary circuit through experimental and numerical investigations, and thus, improve the safety of the nuclear power plant. The obtained results during 2015-2016 are divided into four groups. Firstly, miniature sized fracture toughness specimens were validated and used to determine ductile-to-brittle temperature, T_0, of Barsebäck reactor pressure vessel weld. Secondly, related to dissimilar metal welds, inlay and overlay welding was investigated in cases of weld repair. Thirdly, two types of dissimilar metal welds were characterised. Tearing resistance was lowest at the fusion boundary and conventional specimens produce conservative results. The results show the effect of crack path on tearing resistance. Finally, based on experimental characterisation on dissimilar metal welds, a numerical model was built, and the tearing resistance and crack path was simulated. The tearing resistance gives a good fit to the experimental results, but to capture the crack path a refined numerical model, with thinner material zones, is required.

MOCCA - Mitigation of cracking through advanced water chemistry
This project focuses on advanced water chemistry tools by which the formation of magnetite particulates in the pressurized water reactor (PWR) feed water line and their deposition into the steam generator (SG) can be minimised, and the resulting localised corrosion mitigated. As one of the main results octadecylamine, ODA, a film forming amine has been shown to reduce carbon steel corrosion rate in PWR secondary side conditions by a factor of x3. The main conclusion from the subproject on lead assisted stress corrosion cracking (PbSCC) is that a combination of a mechanical testing method, in this case the SSRT test method, in situ electrochemical methods and ex situ surface analytical methods are needed to gain further insight into the mechanism of PbSCC. Based on the results, the effect of Pb is to activate the surface so that any local corrosion mode, such as SCC, becomes practically impossible, and general corrosion is observed instead. However, if the environment changes to more oxidising one by as little as 0.1 V, Pb dissolves from the surface possibly as $PbCl^+$, leaving the surface in a semi-passive state and very susceptible to SCC.

THELMA - Thermal ageing and EAC research for plant life management

During the first two years of the THELMA project studies of thermal ageing of Alloy 690, started in the SAFIR 2014 ENVIS project have continued. The results show intergranular carbide precipitation, lattice contraction and increased hardness, indicative of ordering reaction. This may affect the long-term primary water stress corrosion cracking (PWSCC) resistance of pressurized water reactor (PWR) components. The studies on thermal ageing of Type 316L weld metals (similar to those used in EPR reactors) show spinodal decomposition, G-phase formation, decreased impact and fracture toughness and increased crack growth rate in BWR conditions. Nano-indentation and DL-EPR results are in good correlation with the degradation of mechanical properties and microstructural changes, implying that these methods could be used for prediction of plant aged boat samples. Initiation testing on austenitic materials (stainless steel and alloy 182) using tapered specimens show a decreasing apparent threshold stress with decreasing strain rate and a beneficial influence of polished versus ground surface, especially in BWR conditions.

COMRADE - Condition monitoring, thermal and radiation degradation of polymers inside NPP containments

Polymer based materials are used in wide range of applications inside nuclear power plant containments e.g. cables, sealants, paint coatings, lubricants and greases. During their designed life time and accidental scenarios they are exposed to stressors such as heat and ionizing radiation. COMRADE was developed to gain better understanding in setting acceptance criterion for polymer components, search for representative components fo ageing studies, studying ageing effects on polymers experimentally and by means of computational material science. After one year of studies, premilinary acceptance criterion has been set for EPDM o-rings, which will be defined in more detail during future studies and broadened to include different materials and o-ring sizes. Attempt towards acquiring “in service” materials for ageing studies has been boardened to comprehend also currently running nuclear power plants. As part of modelling work it was concluded that modelling tools for the ageing of semi-crystalline polymers is currently lacking and future work is focused on that. The synergistic effects of radiation and heat were evaluated on EPDM and CSM and it seems that during a DBA, the radiation is more detrimental for EPDM than heat, as increasing temperature (up to 125°C) can hinder the degradation when simultaneous irradiation treatment is applied. For oxidation profile measurements it was concluded that ToF SIMS would be the most suitable technique out of two others tested. Finally, different semi-empirical methods were identified which could be used in evaluation of severity of dose rate effect if sufficient experimental data excisted.

WANDA - Non-destructive examination of NPP primary circuit components and concrete infrastructure

Addressing the Long Term Operation of nuclear power plants (NPP) beyond 40 years (and even beyond 60 years) will undoubtedly increase the occurrence and severity of known forms of deterioration. Therefore, there is a need to develop tools and methodologies that can maintain the safety while extending the operating lifetimes of NPP reliably. In WANDA that is the objective for the research.

Phased array ultrasonic testing on fatigue cracks with qualified procedures confirms that signal-to-noise -ratio between same size defects can vary a lot depending on the defect type and material anisotropy effect on the wave propagation. Simulation is a tool for NDT research at the present. That is used in WANDA project for SG magnetite deposit studies as well as for ultrasonic sound propagation studies.

New approach to determine POD curve with limited physical flaws was developed using a scanned UT-data from single flawed sample that was digitally altered and extended to provide unlimited variety of data files for POD determination. Inspectors were able to test capability of finding flaws and generate estimated POD curves with this process.

Multiple comparative NDE studies on reinforced concrete structures showed the promise of various techniques in evaluating concrete degradation, providing the basis of the conceptual designs for this study. A preliminary assessment of the basic restrictions and potential complexities involved with construction of large reinforced concrete mock-up has been conducted with a focus on defining the general requirements for the construction of the mock-up.

Research area “RG6 Research Infrastructure”:

JHR - Jules Horowitz Reactor - JHR collaboration & Melodie follow-up

Jules Horowitz Reactor (JHR), a new European material testing reactor (MTR), is currently under construction at CEA Cadarache research centre in France. The JHR consortium has set up three working groups (WG) to determine experimental needs and plan future experiments. After gathering information on the topics of interest for the first experiments from the JHR consortium members, and creating the ranking grids for the selection of the topics, the WGs agreed on fuel and material irradiation experiments, which would become the first experiments planned and performed by the international JHR consortium. The position paper, which describes these pre-JHR experiments proposed by the WGs, was drafted and delivered to the JHR governing board in 2016. The Melodie, Mechanical Loading Device for Irradiation Experiments, delivered to CEA in 2012 as a part of the Finnish in-kind contribution to the international JHR project, is a device for the study of the irradiation creep of a Zircaloy-4 fuel cladding tube specimen. The instrumented test device has the capability to control the biaxial loading and to measure the biaxial strain of the specimen online. The Melodie in-core experiment started in May 2015 and lasted for six reactor cycles. The results of the in-core experiment have not been fully analysed yet, but some initial results are available. The LVDT5, measuring the axial strain, produced consistent low-noise data in just one week, making it possible to analyse the value of the axial creep strain. The behaviour of the loading frame, the gas management system and the data acquisition system was reliable throughout the experiment, which indicates the potential of the technology considering future experiments.

INFRAL - Development of thermal-hydraulic infrastructure at LUT

The aim of the INFRAL project is to develop the thermal hydraulic measurement infrastructure of the LUT nuclear safety research laboratory, to secure the operability of the existing test facilities and to launch a study on the new large-scale integral test facility. During 2015–2016, the so-called advanced measurement techniques, i.e. particle image velocimetry (PIV), wire-mesh sensors (WMSs) and high-speed cameras (HSCs) have been versatilely used in different research projects. The expertise on these measurement systems, as well as on the related data processing procedures, has taken big leaps forward. The study on the new modular integral test facility was launched in 2016. The survey of the research based requirements for the new test facility was conducted on the national level to ensure that the needs of Finnish stakeholders will be fulfilled. In addition, international trends and needs for the thermal hydraulic experimental research were studied to enable participation to various joint international projects in the future. International co-operation with other top-level universities and research institutes has continued, and LUT has formed valuable connections to institutes such as ETH Zurich, Paul Scherrer Institute and University of Michigan.

RADLAB - Radiological laboratory commissioning 2016

The RADLAB project executes the renewal of the radiological research infrastructure hosted by VTT, embodied in the new VTT Centre for Nuclear Safety. In the first half of the SAFIR 2018 programme, the RADLAB project has overseen the design and fabrication of new hot cells for testing and characterization of activated reactor structural materials. It has also executed the procurement of key hot laboratory equipment, the design, fabrication and installation of self-built research facilities, the design, fabrication and installation of materials handling and storage facilities, and contributed to the full laboratory infrastructure commissioning and ramp-up of operations for both reactor safety and final repository research. The hot cells have been designed and manufactured on a contract with Isotope Technologies Dresden, GmbH, all the main key devices have been procured, and the radiological operation permit for the radiochemistry and microscopy facilities has been granted. The installation and commissioning of the new hot cells is on track for 2017.