SAFIR2018 FINAL SEMINAR AND WEBINAR 21.–22.3.2019

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

Venue: Hanaholmen Swedish-Finnish Cultural Centre, Hanasaarenranta 5, Espoo (link)

Day 1: 21.3.2019

Chairs: Jorma Aurela, MEAE & Janne Peltonen, Fennovoima

8:30 – 9:00	**Registration**
9:00 – 9:40	**Opening session: Liisa Heikinheimo (MEAE), Petteri Tiippana (STUK), Jari Hämäläinen (SAFIR2018)**
9:40 – 10:40	**Research area “RG1 Automation, Organisation and Human Factors”**
9:40 – 10:00	*Electric systems and safety in Finnish NPP – ESSI (VTT, Aalto, Risk Pilot),* Seppo Hänninen
10:00 – 10:20	*Management principles and safety culture in complex projects – MAPS (VTT, Aalto, University of Oulu),* Nadezhda Gotcheva
10:20 – 10:30	Poster: *Crafting operational resilience in nuclear domain – CORE (VTT, FIOH),* Jari Laarni
10:30 – 10:40	Poster: *Practical applications and further development of overall safety concept – ORSAPP (LUT, VTT),* Marja Ylönen
10:40 – 11:10	**Coffee break**
11:10 – 11:30	**Research area “RG1 Automation, Organisation and Human Factors” (cont.)**
11:10 – 11:30	*Integrated safety assessment and justification of nuclear power plant automation – SAUNA (VTT, Aalto, FISMA, Risk Pilot),* Antti Pakonen
11:30 – 12:30	**Research area “RG2 Severe Accidents and Risk Analysis”**
11:30 – 11:50	*Extreme weather and nuclear power plants – EXWE (FMI),* Kirsti Jylhä
11:50 – 12:10	*Probabilistic risk assessment method development and applications – PRAMEA (VTT, Aalto, Risk Pilot),* Terhi Kling
12:10 – 12:30	*Safety of new reactor technologies – GENXFIN (VTT),* Tuomo Sevón
12:30 – 13:30	**Lunch**
Chairs: Matti Kattainen, Fortum & Pekka Viitanen, Fennovoima	
13:30 – 14:20	**Research area “RG2 Severe Accidents and Risk Analysis” (cont.)**
13:30 – 13:50	*Fire risk evaluation and defence-in-depth – FIRED (VTT, Aalto),* Jukka Vaari
13:50 – 14:00	Poster: *Comprehensive analysis of severe accidents – CASA (VTT),* Anna Korpinen
14:00 – 14:10	Poster: *Chemistry and transport of fission products – CATFIS (VTT),* Teemu Kärkelä
14:10 – 14:20	Poster: *Experimental and numerical methods for external event assessment improving safety – ERNEST (VTT), Ari Vepsä*
14:20 – 14:40	**Research area “RG3 Reactor and Fuel”**
14:20 – 14:30	Poster: *Development of a Monte Carlo based calculation sequence for reactor core safety analyses – MONSOON (VTT*), Jaakko Leppänen
14:30 – 14:40	Poster: *Safety analyses for dynamical events – SADE (VTT),* Elina Syrjälahti
14:40 – 15:40	**Poster session and coffee break** (poster presenters of research areas RG1, RG2 and RG3 shall be available for discussions)
15:40 – 16:40	**Research area “RG3 Reactor and Fuel” (cont.)**
15:40 – 16:00	*Nuclear criticality and safety analyses preparedness at VTT – KATVE (VTT),* Pauli Juutilainen
16:00 – 16:20	*Physics and chemistry of nuclear fuel – PANCHO (VTT),* Henri Loukusa
16:20 – 16:40	*Uncertainty and sensitivity analyses for reactor safety – USVA (VTT, Aalto),* Asko Arkoma
Chair: Jorma Aurela, MEAE	
16:40 – 17:20	**Invited presentations**
16:40 – 17:00	*Benefits and challenges in participation in international safety research*, Cyril Pinel, Director of International Affairs, IRSN
17:00 – 17:20	*SSM:s view on how to secure nuclear competence in Sweden,* Anneli Hällgren, Director, Development Department, SSM
17:20 – 18:30	**The role of research in the development of safety – Panel discussion with short presentations**
17:20 – 17:30	Hans Wanner, Director General of ENSI, Chair of WENRA (Western European Nuclear Regulators’ Association)
17:30 –	Short presentations by other panelists
– 18:30	Discussion – Hans Wanner, WENRA; Per Seltborg, Research Manager, SSM; Cyril Pinel, IRSN; Petra Lundström, Fortum; Marjo Mustonen, TVO; Timo Okkonen, Fennovoima; Liisa Heikinheimo, MEAE; Petteri Tiippana, STUK
18:30 –	**Buffet dinner**

SAFIR2018

<u>Day 2: 22.3.2019</u>

Chairs: Antti Tarkiainen, TVO & Eero Vesaoja, Fortum

8:30	**Opening**
8:30 – 9:30	**Research area “RG4 Thermal hydraulics”**
8:30 – 8:50	*Comprehensive and systematic validation of independent safety analysis tools – COVA (VTT),* Seppo Hillberg
8:50 – 9:10	*Integral and separate effects tests on thermal-hydraulic problems in reactors – INTEGRA (LUT),* Vesa Riikonen
9:10 – 9:20	Poster: *Couplings and instabilities in reactor systems – INSTAB (LUT),* Markku Puustinen
9:20 – 9:30	Poster: *Development and Validation of CFD Methods for Nuclear Reactor Safety Assessment – NURESA (VTT, Aalto, LUT),* Timo Pättikangas
9:30 – 10:15	**Invited presentations – Topical safety improvements in Finnish nuclear power plants**
9:30 – 9:45	Ari-Pekka Kirkinen, Fortum
9:45 – 10:00	Mikko Lemmetty, TVO
10:00 – 10:15	Hannu Tuulensuu, Fennovoima
10:15 – 10:45	**Coffee break**
10:45 – 12:15	**Research area “RG5 Structural Integrity”**
10:45 – 11:05	*Condition monitoring, thermal and radiation degradation of polymers inside NPP containments – COMRADE (VTT, SP),* Konsta Sipilä
11:05 – 11:25	*Analysis of fatigue and other cumulative ageing to extend lifetime – FOUND (VTT, Aalto),* Juha Kuutti
11:25 – 11:45	*Long term operation aspects of structural integrity – LOST (VTT),* Sebastian Lindqvist
11:45 – 12:05	*Non-destructive examination of NPP primary circuit components and concrete infrastructure – WANDA (VTT, Aalto),* Tuomas Koskinen
12:05 – 12:15	Poster: *Mitigation of cracking through advanced water chemistry – MOCCA (VTT),* Timo Saario
12:15 – 13:15	**Lunch**
Chairs: Tomi Routamo, STUK & Paul Smeekes, TVO	
13:15 – 13:35	**Research area “RG5 Structural Integrity” (cont.)**
13:15 – 13:25	Poster: *Thermal ageing and EAC research for plant life management – THELMA (VTT, Aalto),* Ulla Ehrnstén
13:25 – 13:35	Poster: *Evolving the Fennoscandian GMPEs – EVOGY (VTT, ISUH),* Ludovic Fülöp
13:35 – 14:05	**Research area “RG6 Research Infrastructure”**
13:35 – 13:55	*Barsebäck RPV material used for true evaluation of embrittlement – BRUTE (VTT),* Ulla Ehrnstén
13:55 – 14:05	Poster: *Jules Horowitz Reactor - JHR collaboration & Melodie follow-up – JHR (VTT),* Ville Tulkki
14:05 – 15:05	**Poster session and coffee break** (poster presenters of research areas RG4, RG5 and RG6 shall be available for discussions)
15:05 – 16:00	**Research area “RG6 Research Infrastructure” (cont.)**
15:05 – 15:25	*Development of thermal-hydraulic infrastructure at LUT – INFRAL (LUT),* Joonas Telkkä
15:25 – 16:00	*Radiological laboratory commissioning – RADLAB (VTT),* Wade Karlsen
16:00 – 16:30	**Final discussion**
16:00 – 16:30	Final discussion moderated by the chairperson of the SAFIR2018 Management Board Marja-Leena Järvinen
16:30	**Closing**

<u>Poster sessions:</u>

Research area “RG1 Automation, Organisation and Human Factors”

1. Crafting operational resilience in nuclear domain – CORE (VTT, FIOH), Jari Laarni
2. Practical applications and further development of overall safety concept – ORSAPP (LUT, VTT), Juhani Vihavainen

Research area “RG2 Severe Accidents and Risk Analysis”

3. Comprehensive analysis of severe accidents – CASA (VTT), Anna Korpinen
4. Chemistry and transport of fission products – CATFIS (VTT), Teemu Kärkelä
5. Experimental and numerical methods for external event assessment improving safety – ERNEST (VTT), Ari Vepsä

Research area “RG3 Reactor and Fuel”

6. Development of a Monte Carlo based calculation sequence for reactor core safety analyses – MONSOON (VTT), Jaakko Leppänen
7. Safety analyses for dynamical events – SADE (VTT), Elina Syrjälahti

Research area “RG4 Thermal hydraulics”

8. Couplings and instabilities in reactor systems – INSTAB (LUT), Markku Puustinen
9. Development and Validation of CFD Methods for Nuclear Reactor Safety Assessment – NURESA (VTT, Aalto, LUT), Timo Pättikangas

Research area “RG5 Structural Integrity”

10. Mitigation of cracking through advanced water chemistry – MOCCA (VTT), Timo Saario
11. Thermal ageing and EAC research for plant life management – THELMA (VTT, Aalto), Ulla Ehrnstén
12. Evolving the Fennoscandian GMPEs – EVOGY (VTT, ISUH), Ludovic Fülöp

Research area “RG6 Research Infrastructure”

13. Jules Horowitz Reactor – JHR collaboration & Melodie follow-up – JHR (VTT), Ville Tulkki

Project abstracts

RG1 Automation, Organisation and Human Factors

Electric systems and safety in Finnish NPP – ESSI

The project researched phenomena, impacts and mitigation methods for possible common cause faults in electrical systems caused by open phase conditions (OPC), large lightning strikes and flexible operation of nuclear power plant (NPP). Unbalances of voltages and currents and temperature evolution in motors were measured with different transformer connections in different OPC cases. Effects of OPCs on NPP electrical components and the related protection of components were mapped. Simulation models were developed to study the lighting surges. Surge arrestor, capacitors and external conductive grounding cables parallel to the protected signal cables are very effective at damping fast transient overvoltages. Areas of concern and risks for flexible operation of NPP are thermal system, turbine, control room and personnel, financial profitability and grid stability in low inertia case. A risk analytic approach to assess options for flexible operation was outlined.

Management principles and safety culture in complex projects – MAPS

MAPS project aimed at enhancing nuclear safety by supporting high quality execution of complex nuclear industry projects. The results advance knowledge on key dimensions of governance for safety in inter-organizational project networks and suggest a novel approach for self-assessment based on the governance framework. Insights from the empirical case studies point to importance of enahancing non-technical capabilities, such as communication and collaboration, and aligning fragmented viewpoints. Leading methodical change of safety culture in dynamic and networked organizations requires a systemic approach. System dynamics modelling resulted in an interactive simulation game to support decision-makers in gaining a holistic perspective on patterns of dynamics interrelations in project contexts.

Crafting operational resilience – CORE

The CORE project aimed 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. More specifically, we 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, was developed to analyse human contribution to nuclear safety.

Institutional strength-in-depth in the context of decommissioning and learning from incidents – ORSAPP

The goal of the study is to examine the robustness of Finnish nuclear community in the context of decommissioning and robustness of license holder in the context of incident investigations. The concept of institutional strength-in-depth, introduced by the IAEA (2017) refers to the openness, transparency and questioning attitude, in core organizations in the nuclear sector. The concept is a reference point when examining the robustness in the inter-organizational and organizational contexts. The findings of decommissioning case show that Finnish nuclear community is relatively robust as regards institutional strength-in-depth and its characteristics, namely openness and transparency. However, the stakeholders' respect of each others' roles and responsibilities may hinder the questioning attitude, another characteristic of institutional strength-in-depth. Furthermore, there is no comparison whether VYR funding is sufficient. In order to enhance the robustness of nuclear community in the phase of decommissioning, it would be important to develop risk management towards taking better into account qualitative risks and critical points, such as organizational and motivational aspects. Incident reports show that the instrumental rationality is licensee organization's principal way to handle and govern incidents and learn from them. Accimaps and use of pre-job briefings can be good tools but they remain inadequate to address broader social and organizational context related issues. Therefore, in order to improve the handling of and learning from incidents, and particularly the handling of communication and organizational factors related aspects, adoption of reflexive responsive rationality is recommended.

Integrated safety assessment and justification of nuclear power plant automation – SAUNA

Our general objective in SAUNA (2015–2018) has been to develop integrated methods and tools for safety assessment and transparent safety demonstration of nuclear power plant instrumentation and control (I&C) systems. Due to the multidisciplinary nature of the nuclear power plant as a whole, I&C safety assessment calls for an overall safety point-of-view. We have worked towards that goal by 1) specifying model-based Systems Engineering approaches for I&C, 2) developing concepts and models for analysing Defence-in-Depth issues in I&C architectures, 3) integrating existing methods and finding novel tools for analysing overall safety (with particular focus on formal verification), and 4) developing model-based ways for attesting conformity in the I&C qualification process.

RG2 Severe Accidents and Risk Analysis

Extreme weather and nuclear power plants – EXWE

The general objective of EXWE was to give better estimates of probabilities of extreme geophysical events that affect the design principles of nuclear power plants (NPPs) and may pose external threats to the plants. Extreme incidents in weather, sea level events and space weather, as well as atmospheric dispersion were considered. Various observations, modelling and machine learning approaches were utilized. A significant positive trend in Finland over the past 40 years was detected in a traditionally-used predictor of thunderstorms. If in any of the three NPP sites, the probability of severe freezing rain might increase in Hanhikivi in the future. Simulations of four past coastal snowfall cases with a numerical weather prediction system revealed similarities in their patterns. Flow patterns in summertime sea-breeze circulation and their impacts on atmospheric dispersion of potential releases were also realistically simulated, provided that a sufficiently high resolution is used in the meteorological and dispersion simulations. Improved coastal flooding risk estimates were made by examining the joint effect of high sea level and wind waves. The availability of local wave measurements is a limiting factor in calculating accurate flood risk estimates that take into account the effect of waves. Other sea level research topics included meteotsunamis, sea level and wave model validation as well as mean sea level scenarios on the Finnish coast over the 21st century.

Probabilistic risk assessment method development and applications - PRAMEA

The goal of PRAMEA project has been to develop methods and use of probabilistic risk analysis (PRA), covering most of its currently topical subfields.
A literature survey concerning human reliability analysis (HRA) in advanced control rooms (CR) revealed that there are significant differences in human errors and performance shaping factors between conventional and advanced CRs. Preparation of an IAEA safety report on HRA has been participated. A new framework for the conduct of site level PRA has been created in co-operation with Nordic partners. Dynamic containment event tree modelling techniques have been studied and guidance provided for performing uncertainty analysis in that context. Substantial new features have been implemented to PRA software FinPSA, including integration of PRA levels 1 and 2, a risk integrator for level 2, and a task file for time-dependent analysis. In level 3 PRA, a guidance document has been written in cooperation with Nordic partners, among other things. Novel methods for conducting risk management actions cost-efficiently and in a risk-informed manner, and a method for selecting mitigation actions against cyber threats to electric power systems have been developed.

Safety of new reactor technologies – GENXFIN

The GENXFIN project investigated safety issues of advanced reactor concepts, mainly SMRs. Various international forums and working groups on advanced reactor concepts were participated and the information was distributed to the SAFIR reference group. The project organized a national SMR seminar.
The licensability of the passive safety systems of the NuScale SMR design was investigated. The key issues are probably the independency of the defense in depth levels, and the severe accident management systems. Doses caused by a hypothetical radioactive release from a NuScale reactor were calculated. With conservative release estimates, sheltering could be needed within 5–12 km of the plant, and evacuation could be needed within 2–8 km.
Modeling methods of passive safety systems were developed by calculating PANDA isolation condenser experiments with the MELCOR code. When compared with earlier calculations with other codes, MELCOR gave about equally good results as the APROS code and better results than the RELAP5 code. As an in-kind contribution to the GENXFIN project, Fortum shared a report of their APROS modeling work of the NuScale and Yanlong SMRs.

Fire risk evaluation and Defence-in-Depth – FIRED

A significant proportion of the overall core damage risk in nuclear power plants is associated with internal fires. The computational tools that are used for assessing the fire risks have developed significantly over the last ten years: the deterministic analyses are now solely based on CFD, and probabilistic analyses using Monte Carlo –simulation have been carried out. Maintaining the computational infrastructure requires continuous investments, and significant research efforts are still needed to enable predictive engineering simulations of fire spreading. The main themes covered by the FIRED project were fire risks of cables during the plant life cycle, fire Defence-in-Depth, and modelling tool development and validation. In addition, active participation in OECD PRISME 2 –project was continued.

Comprehensive analysis of severe accidents – CASA

Overall understanding of the progress and mitigation of severe accidents was strengthened by simulating the Fukushima accidents. When determining the post-dryout behaviour of the debris beds, the friction model effects notably the level the temperature stabilizes. VTT's modelling framework for hydrogen combustion works reasonably as the maximum pressure is close to the experimental result and the pattern of the flame front propagation is similar enough. MELCOR simulations are in a good agreement with the pool scrubbing experiments for non-soluble aerosols whereas ASTEC results were in a good comparison with the experiments for soluble aerosols. The offsite dispersion and dose assessment codes VALMA and ARANO typically predicts smaller dose values than widely validated MACCS.

Chemistry and transport of fission products – CATFIS

The aim in the CATFIS project during 2015–2018 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 understanding on fission products retention in the water pools of containment due to pool scrubbing was improved. The phenomenon was studied with experiments and simulations. The follow-up of OECD/NEA STEM-2 and BIP-3 projects was carried out.

Experimental and numerical methods for external event assessment improving safety – ERNEST

Aircraft impact is one of the external threats that nuclear power plant structures might undergo. WTC terrorist attacks showed that also deliberate crashes with large passenger aircrafts are possible. These impacts and their consequences can be analyzed with sophisticated computational tools. However, in order to be reliable, these tools must be validated against relevant experimental data. The main purpose of ERNEST project has been to carry out selected medium scale impact tests with reinforced concrete slabs, to develop modelling tools used at VTT for analyzing this type of events and to validate these tools against experimental test data. The tests have been carried out with VTT's own testing apparatus. Development of a concrete material model has been a major part of the modelling tool development. The material model has been implemented into Abaqus finite element code and it enables more realistic behavior of concrete under impact loading. The tools, including the material model, have been validated mainly against test data collected from tests carried out at VTT.

RG3 Reactor and Fuel

Development of a Monte Carlo based calculation sequence for reactor core safety analyses – MONSOON

The MONSOON project continued 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. During the first years the work covered development of methodologies for spatial homogenization, which forms the first part of the traditional multi-stage calculation scheme applied to core physics calculations. The focus was later shifted to the development of a new nodal neutronics code Ants as part of the new Kraken computational framework. The work on Serpent development was started during the previous SAFIR programmes, and the KÄÄRME project in SAFIR2014 provided a proof-of-concept type demonstration for the use of the continuous-energy Monte Carlo method for spatial homogenization. The purpose of the MONSOON project was to proceed from feasibility studies to practical applications. Serpent development is carried out in close collaboration with a large international user community. The code is currently used in more than 200 organizations in 42 countries around the world

Safety analyses for dynamical events – SADE

Capabilities to model transients and accidents have been improved in the SADE project. The focus has been in the VVER modelling, with two parallel development branches: tools for practical safety analyses and more ambitious approach to use CFD-type 3D thermal-hydraulics in reactor dynamics modelling. During this four year project the coupling between VTT's reactor dynamics code HEXTRAN and system code SMABRE was extended with the new internally coupled simulation mode, which can be now used for routine safety analyses. Several improvements have been done to the neutronics modelling and also to the whole safety analyses calculation sequence. Group constants created with Serpent-2 have been taken into use in VTT's reactor dynamics codes that makes it possible to use independent, domestic calculation tools for the all phases of the safety analyses. Breakthrough was achieved when first truly coupled transient simulations were done using CFD code for reactor pressure vessel, the HEXTRAN code for core neutronics and SMABRE for the rest of the reactor coolant system.

Nuclear criticality and safety analyses preparedness at VTT – KATVE

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. The information was used to evaluate the fuel integrity in the cask. 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.

Physics and chemistry of nuclear fuels – PANCHO

PANCHO has been the essential platform for the ongoing development of FINIX fuel behaviour module and its accompanying automatic validation system. FINIX has been published through the OECD/NEA Data Bank, and has been licensed by several institutions around the world. 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. A coupling between SCANAIR and GENFLO codes has been developed in the project. The chemical experiments on nuclear fuel materials aimed to investigate characteristics of initial dissolution of crystalline ThO_2 by adding ^{229}Th tracer into the aqueous phase, and were conducted for total 534 days. The experimental campaign on cladding creep demonstrated that the VTT equipment is capable of performing suitable transient tests which support modelling efforts. Three DSc theses were finalized based on the ongoing cladding creep modelling work, the modelling of fuel behavior in design basis accidents and fuel pellet dissolution experiments. One Master's thesis is to be finalized from the work performed in this project.

Uncertainty and sensitivity analyses for reactor safety – USVA

In USVA project, multidisciplinary uncertainty and sensitivity studies in reactor safety have been conducted. The most important accomplishments of the project include a thorough comparison of statistical sensitivity analysis methods in the context of fuel behavior modelling (Ikonen, 2015, 2016), sensitivity analysis of local uncertainties in a multicode calculation chain of a large break loss of coolant accident (LB-LOCA) (Arkoma and Ikonen, 2016a,b), development of an automated uncertainty analysis calculation system for the CASMO-4 – SIMULATE-3 calculation chain (Pusa, 2016; Pusa and Isotalo, 2017), and initial developments of uncertainty analysis for the group constant generation with Serpent (Valtavirta et al. 2018, Valtavirta 2018). Two publications of USVA were used in a doctoral dissertation (Arkoma, 2018c). A Master's thesis was finalized on the combined uncertainty analysis of coupled neutron transport and fuel behaviour codes (Taavitsainen, 2016).

RG4 Thermal hydraulics

Comprehensive and systematic validation of independent safety analysis tools – COVA

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 was 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 enhanced the expertise in thermal hydraulic area of Generation II and III LWR reactors and included, as an essential part, training of new experts to this relevant area of reactor safety. While main effort was 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 was also used in the analyses.

Integral and separate effects tests on thermal-hydraulic problems in reactors – INTEGRA

The objective of the SAFIR2018 INTEGRA project was to improve the understanding of thermal-hydraulic system behavior by performing integral and separate effects tests with the PWR PACTEL and PASI facilities, in particular regarding the impact of non-condensable gases on core cooling 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. The data is available also for 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 behavior. 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.

Couplings and instabilities in reactor systems – INSTAB

In the INSTAB project, mechanisms and efficiency of mixing with different subsystems in the pressure suppression pool have been investigated experimentally in the PPOOLEX facility. The tests with a model of a safety relief valve sparger verified that mixing of a thermally stratified water pool could happen also through an erosion process in addition to internal circulation, if suitable flow conditions prevail. The wetwell spray tests in PPOOLEX revealed that mixing of a thermally stratified pool with the help of spray injection from above is possible. The separate effect tests conducted in the SEF-POOL facility to study the details of direct contact condensation, while steam is discharged through a model of a sparger, provided valuable data for the development and validation of the EMS and EHS models done by KTH.

Development and validation of CFD methods for nuclear reactor safety assessment – NURESA

Computational Fluid Dynamics (CFD) methods have been developed and validated for the identified most important topics in nuclear reactor safety assessment. International single-phase mixing benchmarks were participated and spray experiments performed at LUT were modelled in co-operation with Swedish partners. Models for the departure from nucleate boiling (DNB) have been developed for the OpenFOAM code and co-simulations of NPP components with CFD code and Apros system code have been performed.

RG5 Structural Integrity

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

In COMRADE project several ageing related issues in NPP environment were studied, including setting acceptance criterion for O-rings, use of real components in ageing studies, combined effects of radiation and heat and dose rate effect. As a result, several end-of-life criteria was suggested for EPDM O-rings based on the functional property of the O-ring. FE-simulations were conducted to estimate how O-rings endured in a realistic environment. Feasable ways to acquire polymeric materials from running plants and plants underdecomissioning were evaluated. MD-simulations were applied in explaining the mechanisms governing the reverse temperature effect. As part of studying the synergistic effects of radiation and heat, two different kind of behaviour were observed with EPDM and Lipalon cable jacket material in combined high temperature-radiation environments. Also, it was shown that the used sequence in ageing treatments do matter in case of EPDM. Finally the use of semi-empirical models were estimated in the use of predicting dose rate effect and it seems that they require lot of experimental data to provide reliable predictions.

Analysis of fatigue and other cumulative ageing to extend lifetime – FOUND

A summary of the results obtained in the SAFIR2018 FOUND project is presented. The project is focused in ageing and failure assessment of NPP components. The main results consists in the developments in evaluating the criticality of cracks in NPP components, a dissertation on the ageing of the BWR RPV and its internals, new experimental research on the primary water affected fatigue, improved methods to assess thermally induced cyclic loads and thermal fatigue, novel probabilistic and risk-informed methods to assess piping failures and developments in assessing the dynamic response of piping systems. In addition, novel research on residual stress measurement techniques and results on the residual stresses in NPP piping components that have been in service for decades are presented.

Long term operation aspects of structural integrity – LOST

The main objective for the project long term operation aspects of structural integrity (LOST) is to develop methods and tools for structural safety analysis of primary circuit components, reactor pressure vessel (RPV) and dissimilar metal welds (DMW). The work packages are divided accordingly. Work package 1 focuses on reactor pressure vessel safety; 1) fast fracture in upper shelf region 2) pre-investigations for BREDA. Work package 2 focuses on dissimilar metal welds; 1) residual stresses 2) materials characterisation and 3) numerical simulations. The following was achieved during the four years; 1) the fracture toughness assessment for DMWs is safer than before, 2) the residual stresses in DMWs, also after repair welding, are better understood, 3) the various conditions leading to fast fracture in the upper shelf region have been identified, 4) a new numerical model for predicting ductile crack growth was developed 5) also improvements in RPV surveillance methods were achieved. The work continues in SAFIR2022, AMOS project, to improve analysis and characterisation methods for safety class 1 components.

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

Passive components in nuclear power plants (NPP)s are monitored through in-service inspections (ISI)s. These passive components consists for example primary cirquit piping, steam generator and safety related concrete infrastructure such as the concrete containment. ISIs are heavily related to monitoring the condition of aging NPPs. WANDA has addressed these issues by focusing on three important aspects of proactive ageing management: early detection of deterioration, monitoring of deterioration, and application of prognostics for the estimation of remaining service life. NDE is one of the recommended tools for the early detection of deterioration of NPP materials.
Work in WANDA was divided into two work packages, in work package one the focus was on early detection deterioration and inspection reliability. These contained research on investigating plausible ways to detect material deteroation and research on efficient ways to evaluate probability of detection curves for ISI. In work package two the focus was on concrete infrastructure. Mainly the goal was to bring concrete NDE research on a par with metal NDE, thus doing foundational work on first concrete containtment mock-up for research purposes and constructing the mock-up itself.

Mitigation of cracking through advanced water chemistry – MOCCA

Corrosion problems in the PWR secondary circuit are mostly related to deposition of magnetite particles originating in the feed water line into steam generator (SG) and enrichment of impurities into crevices under these deposits. In this work, the effect of alternative water chemistry regimes on the feed water line corrosion rate, the tendency of magnetite to deposit into the SG and on the effect of lead (Pb) as a potentially detrimental impurity within the deposits has been studied.

Thermal ageing and EAC research for plant life management – THELMA

The project THELMA, Thermal ageing and EAC research for plant life management, deals with nuclear materials behaviour in light water reactor environments. The development of thermal ageing of stainless steels, both welds in wrought material and cast material, have been investigated. The increase in hardness of the ferrite phase is a good indicator of ageing, while application of electrochemical measurement method double loop electrochemical potentiokinetic reactivation is not straightforward for materials aged in operation. It is, though, a feasible method when ageing has been performed at higher (~400°C) temperature. Thermal ageing of Alloy 690 cause intergranular carbide precipitation, lattice contraction and increased hardness, indicative of ordering reaction. This may affect the long-term primary water stress corrosion cracking resistance of pressurized water reactor (PWR) components. Detailed microstructural investigations of mechanical test specimens made from Barsebäck 2 pressure vessel steel weld metal show e.g. a correlation between a low fracture toughness and brittle fracture initiation at large secondary particles. The detailed results form a basis for similar investigations to be performed on irradiated reactor pressure vessel materials. The research increase the knowledge on materials behaviour in nuclear power plant environments which is needed to assist the safety authority STUK, licensees and other stakeholders in questions concerning plant life management, material degradation assessment and failure analysis. Advanced knowledge must exist in advance, and cannot be built in e.g. failure analysis work, which typically require a short through-put time.

Evolving the Fennoscandian GMPEs – EVOGY

Evolving the Fennoscandian GMPEs (EVOGY) project was targeted at proposing an updated ground motion prediction equation (GMPE) for probabilistic seismic- hazard analyses of Finnish nuclear installations. We collected and archived the available recordings of earthquakes in Fennoscandia and created a database with all spectral components important for engineering evaluation. For developing the GMPE, we used the backbone curves of the G16 equation proposed by Graizer (2016) for central and eastern North America. We adjusted the peak ground acceleration (PGA) prediction of G16 to cover lower magnitudes and very hard-rock conditions of Fennoscandia, but keep it unchanged for magnitudes above M_w4. We adjust the normalized spectral shape prediction of G16 using combined Fennoscandian and NGA-East data for very hard rock. We evaluated the mean prediction and error using the calibration data, and compared the adjusted GMPE to the subset of hard-rock recordings from the NGA-East database. We used a set of synthetic ground motions created using a hybrid modeling method to confirm the prediction in the near field. We conclude that the adjusted G16 formulation is adequate for predicting ground motions in Fennoscandia. Due to the compatibility with the original G16 backbone curve and additional comparisons, we estimate that the validity of the proposed formulation is up to the range of $2.0 \le M_w \le 6.5$ and $0 \le R_{rup} \le 300$km for the hard-rock conditions.

RG6 Research Infrastructure

Barsebäck RPV material used for true evaluation of embrittlement – BRUTE

The BRUTE project has two corner stones, i.e. 1) pioneering the new hot cell infrastructure at the Centre for Nuclear Safety, VTT and 2) performing mechanical testing and microstructural investigations of Barsebäck 2 reactor pressure vessel (RPV) material. BRUTE is part of a larger entity, the Barsebäck REsearch&Development Arena, BREDA. BRUTE started in 2018 with preparatory work for mechanical testing and microstructural investigations, which will be performed in the SAFIR2022 BRUTE project. One of the main objectives of the BREDA and BRUTE projects is to investigate the correlation between fracture toughness properties of the RPV and the surveillance specimens. In the BREDA project, cylindrical trepans á ~40 kg, with a diameter of ~200 mm were extracted from B2 RPV beltline welds, subjected to a neutron irradiation, as well as RPV head welds, which are not subjected to irradiation, but to thermal loading. In BRUTE2018, the rationale for the investigations were fine-tuned, plans for microstructural investigations were made and a preliminary test matrix, comprising of more than 1500 mechanical test specimens was developed. Qualification and verification of test methods is of very high importance to obtain reliable results with small scatter. Most of the selected testing techniques for mechanical tests and microstructural investigation were validated using the new infrastructure, in co-operation with the SAFIR2018 RADLAB project. The seminar "Reactor pressure vessel embrittlement seminar for the BREDA – BRUTE project" gathered >30 participants from Finland and Sweden, and the state the art was presented, and launched the formation of a Nordic knowledge pool on RPV embrittlement including three doctoral students in three different universities.

JHR Collaboration & Melodie follow-up – JHR

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, with preparatory work on-going afterwards. 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 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.

Development of thermal-hydraulic infrastructure at LUT – INFRAL

The aim of the INFRAL project was to develop the thermal hydraulic measurement infrastructure of the LUT University 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 the SAFIR2018 research programme, the advanced measurement techniques, i.e. particle image velocimetry (PIV), wire-mesh sensors (WMSs) and high-speed cameras (HSCs) were versatilely used in different research projects. The expertise on these measurement systems, as well as on the related data processing procedures, took big leaps forward. The study on the new modular integral test facility, titled MOTEL (MOdular TEst Loop), 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. Further, the modularity based requirements for the MOTEL facility were studied, and the first configuration of the test facility was introduced. International co-operation with other top-level universities and research institutes continued during SAFIR2018, and thus LUT has formed valuable connections to research institutes such as Paul Scherrer Institute, University of Michigan and Helmholtz-Zentrum Dresden-Rossendorf.

Radiological laboratory commissioning – RADLAB

The RADLAB project executes the renewal of the radiological research infrastructure hosted by VTT, embodied in the new VTT Centre for Nuclear Safety. At the core of the new facility are the hot cells for testing and characterization of activated reactor structural materials in support of safe, long-term nuclear power plant operation. In the first half of the SAFIR2018 program (see Interim Report), the project oversaw the design and fabrication of the new hot cells, while in the second half of the program the focus has been on the on-site installation and bringing them into full operation. Simultaneously the project has executed the procurement of key hot laboratory equipment, the purchase cost of which has been supported by the investment aid mechanism financed through the complementary RADINFRA project. Additional activities have included the design, fabrication and installation of self-built research facilities and materials and waste-handling and storage facilities, as well as the full laboratory infrastructure commissioning and ramp-up of operations for both reactor safety and final repository research.