RESEARCH REPORT

VTT-R-00618-19

SAFIR2018 Annual Report 2018

Authors: Jari Hämäläinen & Vesa Suolanen

Confidentiality: Public

Report's title	
SAFIR2018 Annual Report 2018	
Customer, contact person, address SAFIR2018/VYR	**Order reference**
Project name Administration of the research programme SAFIR2018	**Project number/Short name** 118214_Admire_18_2018
Author(s) Jari Hämäläinen & Vesa Suolanen	Pages 118 + 50 app.
Keywords Nuclear safety, research programmes, SAFIR2018	**Report identification code** VTT-R-00618-19

Summary

The mission of the National Nuclear Power Plant Safety Research programme 2015-2018 (SAFIR2018) is derived from the stipulations of the Finnish Nuclear Energy Act. The programme is continuation to a series of earlier national nuclear power plant safety research programmes that have proven their worth in maintaining and developing expertise.

SAFIR2018 Management Board is responsible for steering and planning of the research programme and consists of the representatives of the Radiation and Nuclear Safety Authority (STUK), the Ministry of Economic Affairs and Employment (MEAE), Fennovoima Oy, Fortum, Teollisuuden Voima Oyj (TVO), Technical Research Centre of Finland Ltd (VTT), Lappeenranta University of Technology (LUT), Aalto University (Aalto), and Swedish Radiation Safety Authority (SSM).

The actual volume of the SAFIR2018 programme in 2018 was 7,2 M€ and 49 person years. Main funding organisations in 2018 were the Finnish State Waste Management Fund (VYR) with 4,3 M€ and VTT with 1,5 M€. The programme was divided into three research areas and in 2018 research was carried out in 32 projects.

This report provides a summary of the results of the individual projects and overall financial and administrative issues. Summaries of project publications, international cooperation, academic degrees and international travels are presented in the Appendices.

Confidentiality	Public

Espoo 14.6.2019

Written by	Reviewed by	Accepted by
Jari Hämäläinen Programme Director	Petri Kinnunen Research Manager	Satu Helynen Vice President, Nuclear Safety

VTT's contact address
VTT, P.O. Box 1000, FI-02044 VTT, Finland

Distribution (customer and VTT)

SAFIR2018 website (http://safir2018.vtt.fi/), VTT archive

Contents

Contents 2

1. Introduction 4

2. Main results of the research projects in 2018 6

2.1 Plant safety and systems engineering 9
2.1.1 CORE - Crafting operational resilience in nuclear domain 9
2.1.2 EXWE - Extreme weather and nuclear power plants 12
2.1.3 MAPS - Management principles and safety culture in complex projects 16
2.1.4 PRAMEA - Probabilistic risk assessment method development and applications 19
2.1.5 SAUNA - Integrated safety assessment and justification of nuclear power plant automation 22
2.1.6 GENXFIN - Safety of new reactor technologies 26
2.1.7 ESSI - Electric systems and safety in Finnish NPP 30
2.1.8 ORSAPP - Practical applications and further development of Overall Safety Concept 35
2.2 Reactor safety 37
2.2.1 CASA - Comprehensive analysis of severe accidents 38
2.2.2 CATFIS - Chemistry and transport of fission products 41
2.2.3 COVA - Comprehensive and systematic validation of independent safety analysis tools 45
2.2.4 INSTAB - Couplings and instabilities in reactor systems 47
2.2.5 INTEGRA - Integral and separate effects tests on thermal-hydraulic problems in reactors 51
2.2.6 KATVE - Nuclear criticality and safety analyses preparedness at VTT 54
2.2.7 MONSOON - Development of a Monte Carlo based calculation sequence for reactor core safety analyses 58
2.2.8 NURESA - Development and validation of CFD methods for nuclear reactor safety assessment 60
2.2.9 PANCHO - Physics and chemistry of nuclear fuel 61
2.2.10 SADE - Safety analyses for dynamical events 65
2.2.11 USVA - Uncertainty and sensitivity analyses for reactor safety 68
2.3 Structural safety and materials 71
2.3.1 ERNEST - Experimental and numerical methods for external event assessment improving safety 71
2.3.2 FIRED - Fire risk evaluation and Defence-in-Depth 72
2.3.3 FOUND - Analysis of fatigue and other cumulative ageing to extend lifetime 75
2.3.4 LOST - Long term operation aspects of structural integrity 80
2.3.5 MOCCA - Mitigation of cracking through advanced water chemistry 82
2.3.6 THELMA - Thermal ageing and EAC research for plant life management 87

2.3.7 WANDA - Non-destructive examination of NPP primary circuit components and concrete infrastructure 90
2.3.8 COMRADE - Condition monitoring, thermal and radiation degradation of polymers inside NPP containments........ 91
2.3.9 EVOGY - Evolving the Fennoscandian GMPEs 95
2.4 Research infrastructure........ 99
2.4.1 INFRAL - Development of thermal-hydraulic infrastructure at LUT........ 99
2.4.2 JHR - JHR collaboration & Melodie follow-up........ 104
2.4.3 RADLAB - Radiological laboratory commissioning........ 104
2.4.4 BRUTE - Barsebäck RPV material used for true evaluation of embrittlement........ 109
3. Financial and statistical information........ 112
4. Programme management 117
References........ 118

Appendices

Appendix 1 Publications in the projects in 2018

Appendix 2 Participation in international projects and networks in 2018

Appendix 3 Academic degrees obtained in the projects in 2018

Appendix 4 International travels in the projects in 2018

1. Introduction

In accordance with Chapter 7a of the Finnish Nuclear Energy Act enacted in 2004, the objective of the National Nuclear Power Plant Safety Research programme 2015-2018 SAFIR2018 is to ensure that should new matters related to the safe use of nuclear power plants arise, the authorities possess sufficient technical expertise and other competence required for rapidly determining the significance of the matters. High scientific quality is required of the research projects in the programme. The results must also be available for publication.

The SAFIR2018 programme's planning group, nominated by the Ministry of Economic Affairs and Employment in March 2014, stated the following mission for national nuclear safety programmes:

> *National nuclear safety research develops and creates expertise, experimental facilities as well as computational and assessment methods for solving future safety issues.*

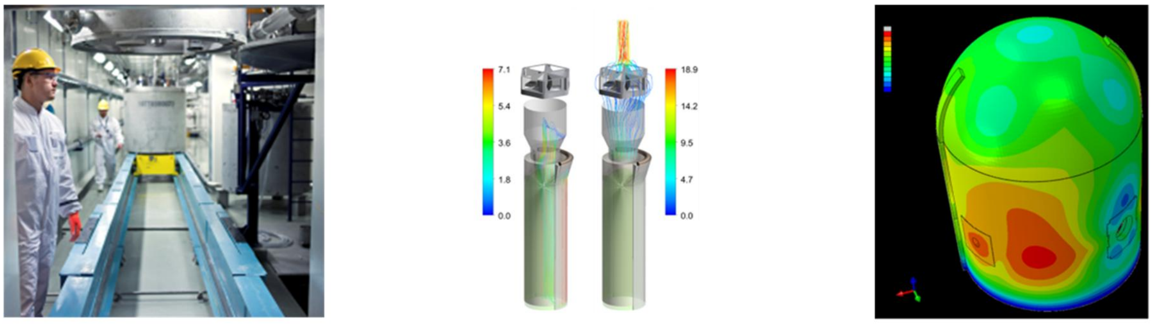
The vision of SAFIR2018 was defined as follows:

> *The SAFIR2018 research community is a vigilant, internationally recognised and strongly networked competence pool that carries out research on topics relevant to the safety of Finnish nuclear power plants on a high scientific level and with modern methods and experimental facilities.*

The Framework Plan [1] describes the research to be carried out in SAFIR2018. The new programme essentially covered the themes of the preceding SAFIR2014 programme [2].

SAFIR2018 Management Board (MB) was nominated in September 2014. It consists of representatives of the Radiation and Nuclear Safety Authority (STUK), the Ministry of Economic Affairs and Employment (MEAE), Fennovoima Oy, Fortum, Teollisuuden Voima Oyj (TVO), Technical Research Centre of Finland Ltd (VTT), Aalto University (Aalto), Lappeenranta University of Technology (LUT), and the Finnish Funding Agency for Innovation (Tekes). In 2015 the management board was completed with a representative of Swedish Radiation Safety Authority (SSM). A representative of Business Finland was not nominated to the MB (Tekes and Finpro unites as Business Finland in the beginning of 2018).

A public call for research proposals for 2018 was announced on 25 August 2017. After the closure of the call, SAFIR2018 management board, taking into account the evaluations made by the steering groups, prepared a proposal for the MEAE regarding the projects to be funded in 2018. The funding decisions were made by the Finnish State Nuclear Waste Management Fund (VYR) in March 2018. In 2018 the programme consisted of 32 research projects and a project for programme administration.



Plant safety and systems engineering Reactor safety Structural safety and materials

Figure 1.1. SAFIR2018 research areas.

VYR funding is collected from the Finnish utilities Fennovoima Oy, Fortum and Teollisuuden Voima Oyj based on their MWth shares in Finnish nuclear power plants (units in operation, under construction, and in planning phase according to the decisions-in-principle). In addition to VYR, other key organisations operating in the area of nuclear safety also fund the programme.

The planned [3] and actual volumes of the SAFIR2018 programme in 2018 were 7,1 M€ and 7,2 M€, and 45 and 49 person-years, respectively.

This annual report summarises the results of the individual projects (Chapter 2) and provides financial statistics of the research programme (Chapter 3). Administrative issues are summarised in Chapter 4.

Project publications are listed in Appendix 1, information on international co-operation in Appendix 2, list of Academic degrees obtained in Appendix 3, list of international travels in the projects in Appendix 4.

This report has been prepared by the programme director and project co-ordinator in cooperation with the managers and staff of the individual research projects.

2. Main results of the research projects in 2018

The SAFIR2018 research programme is divided into three major research areas:

1. Plant safety and systems engineering
2. Reactor safety
3. Structural safety and materials.

The research areas are presented with more detailed descriptions of their research needs during the programme period 2015-2018 in the SAFIR2018 Framework Plan [1]. The research areas and research needs are based on the knowledge at the time of making the framework plan. The Framework Plan will be updated during the programme period, if necessary.

In 2018, the research was performed in altogether 32 research projects. The total volume of the programme was 7,2 M€ and 49 person years. The research projects in the various research areas with their planned and actual volumes are given in Table 2.1.

Summaries of research project results are given in the following subsections.

Table 2.1. SAFIR2018 projects in 2018.

Research area	Project	Acronym	Organi-sation(s)	Planned costs (k€)	Actual costs (k€)	Planned volume (person months)	Actual volume (person months)
1. Plant safety and systems engineering							
	Crafting operational resilience in nuclear domain	CORE	VTT, FIOH	202,0	198,6	16,3	15,2
	Extreme weather and nuclear power plants	EXWE	FMI	200,0	231,2	19,8	26,8
	Management principles and safety culture in complex projects	MAPS	VTT, Aalto, University of Oulu	160,0	163,4	11,5	11,0
	Probabilistic risk assessment method development and applications	PRAMEA	VTT, Aalto, Risk Pilot	328,0	328,2	25,4	25,6
	Integrated safety assessment and justification of nuclear power plant automation	SAUNA	VTT, Aalto, FISMA, Risk Pilot	354,0	355,8	27,5	29,2
	Safety of new reactor technologies	GENXFIN	VTT	94,0	96,0	5,6	5,8
	Electric systems and safety in Finnish NPP	ESSI	VTT, Aalto	130,0	122,5	11,3	11,9
	Practical applications and further development of Overall Safety Concept	ORSAPP	LUT, VTT	71,0	69,4	5,0	5,0
2. Reactor safety							
	Comprehensive analysis of severe accidents	CASA	VTT	233,0	237,3	14,6	14,2
	Chemistry and transport of fission products	CATFIS	VTT	142,9	144,3	8,0	8,7
	Comprehensive and systematic validation of independent safety analysis tools	COVA	VTT	234,0	235,5	16,5	18,4
	Couplings and instabilities in reactor systems	INSTAB	LUT	140,0	141,1	11,0	12,6
	Integral and separate effects tests on thermal-hydraulic problems in reactors	INTEGRA	LUT	326,0	326,8	21,0	35,3
	Nuclear criticality and safety analyses preparedness at VTT	KATVE	VTT	200,6	187,6	15,4	15,4
	Development of a Monte Carlo based calculation sequence for reactor core safety analyses	MONSOON	VTT	158,0	142,8	11,3	10,0
	Development and validation of CFD methods for nuclear	NURESA	VTT, LUT	203,0	201,5	14,4	15,0

	reactor safety assessment						
	Physics and chemistry of nuclear fuel	PANCHO	VTT	259,0	240,6	19,0	20,2
	Safety analyses for dynamical events	SADE	VTT	114,2	114,4	8,3	7,6
	Uncertainty and sensitivity analyses for reactor safety	USVA	VTT	115,0	108,6	9,0	8,3
3. Structural safety and materials							
	Experimental and numerical methods for external event assessment improving safety	ERNEST	VTT	115,0	114,9	6,0	6,4
	Fire risk evaluation and Defence-in-Depth	FIRED	VTT, Aalto	201,0	203,9	14,0	15,1
	Analysis of fatigue and other cumulative ageing to extend lifetime	FOUND	VTT, Aalto	311,0	313,8	17,9	20,2
	Long term operation aspects of structural integrity	LOST	VTT	276,0	287,3	14,3	18,4
	Mitigation of cracking through advanced water chemistry	MOCCA	VTT	147,0	158,0	8,1	7,9
	Thermal ageing and EAC research for plant life management	THELMA	VTT, Aalto	242,0	246,6	17,0	14,7
	Non-destructive examination of NPP primary circuit components and concrete infrastructure	WANDA	VTT, Aalto	167,1	175,4	10,9	13,7
	Condition monitoring, thermal and radiation degradation of polymers inside NPP containments	COMRADE	VTT, SP	159,0	199,7	7,2	7,1
	Evolving the Fennoscandian GMPEs	EVOGY	VTT, ISUH	164,0	160,5	11,5	15,9
4. Research infrastructure							
	Development of thermal-hydraulic infrastructure at LUT	INFRAL	LUT	286,0	296,0	19,0	22,4
	JHR collaboration & Melodie follow-up	JHR	VTT	29,0	29,0	1,7	1,7
	Radiological laboratory commissioning	RADLAB	VTT	694,0	697,1	44,5	45,4
	Barsebäck RPV material used for true evaluation of embrittlement	BRUTE	VTT	286,0	285,9	16,0	17,0
0. Programme administration							
	SAFIR2018 administration	ADMIRE	VTT	377,0	377,0	12,5	12,8

The costs of ADMIRE are for period 1.1.2018-31.3.2019. The costs include the small projects and value-added tax 24%.

2.1 Plant safety and systems engineering

In 2018 the research area “Plant safety and systems engineering” consisted of eight projects:

1. Crafting operational resilience in nuclear domain (CORE)
2. Extreme weather and nuclear power plants (EXWE)
3. Management principles and safety culture in complex projects (MAPS)
4. Probabilistic risk assessment method development and applications (PRAMEA)
5. Integrated safety assessment and justification of nuclear power plant automation (SAUNA)
6. Safety of new reactor technologies (GENXFIN)
7. Electric systems and safety in Finnish NPP (ESSI)
8. Practical applications and further development of Overall Safety Concept (ORSAPP).

2.1.1 CORE - Crafting operational resilience in nuclear domain

The aim of the CORE project (2015-18) is to improve safe operation of nuclear power plants by developing guidance, training interventions, and other practical solutions that promote resilience for the three general defence levels of prevention, preparation, and consequence management. Regarding prevention, the aim is to support operating personnel to succeed better in challenging work tasks by being more reflexive, engaged, and self-conscious and aware of high-level goals, instead of being solely guided by fixed and predetermined procedures. The aim is also to develop new Human Factors guidelines, models, and tools and interventions that will be tested and examined in simulated test environments and in workshops. Regarding preparation, operating personnel needs generic skills and abilities to master difficult, unfamiliar, and ‘knowledge-intensive’ operational situations. They need skills to cope with excessive acute stress in demanding operational situations. There is also need to collect operating experiences from successful actions and decisions and analyse the lessons learned from these experiences. Regarding consequence management and recovery, it is required that risk is efficiently detected, recognized, interpreted, and communicated so that a collective response is mobilized promptly. Therefore, such methods and tools are needed in crisis management that help stakeholders with different responsibilities to coordinate their actions to achieve a common operational picture.

Specific goals in 2018

WP1 focusses on examining how learning from successes could be included in nuclear operating experience activities. It was found that successes are often less salient and less likely to trigger intentional learning processes than failures. It was also observed similar tendencies in normal situations where a task had been executed as expected: further investigation into how exactly the success was achieved (to create lessons) was often not found motivating – the successes were “business as usual”. Facilitating learning from success is thus likely to require deliberate effort, such as its formal inclusion into existing practices for collecting or analysis purposes. A guideline has been developed proposing eight basic principles on how to learn from successes. The guideline also describes a step-by-step process for implementing it in practice. The purpose of this guideline is to provide insights to practitioners working at safety-critical organizations on how to promote learning from successes and to help develop practical tools to achieve this goal.

Within WP2, a work-based learning method was developed guiding the operators to discuss and reflect their performance in simulator training sessions. The method includes individual and group reflection of the applied work practices – practices are, thus, evaluated. Good practices are disseminated between and within operator crews. The reflection is done in view of the established performance criteria of the plant. As a result on the method introduced, it

was found found that reflection was quite abundant, consisting of more than a half of all discussions. In this sense, self-evaluations were efficient in enhancing operators' reflection: they involved dialogue and reflection among operator crews about their own work practices and capability in emergency situations. Overall, findings portray a vivid discussion among the operator shifts encompassing the following themes: work practices, collaboration, plant dynamics and stress at emergency situations.
Within WP3, a summary of the work done in WP3 has been prepared. The report includes a summary of the work done on multitasking and interruption management, and cognitive heuristics in decision making and troubleshooting. It also includes a theoretical synthesis of WP3 work and provides a set of guidelines regarding multitasking, decision-making and troubleshooting. Expert workers' ability to multitask is a resource that will be useful in a state of emergency. On the other hand, in normal work situations, the amount of work should be adjusted to the optimal level so that people have time to think and anticipate. It is suggested that negative consequences of interruptions can be mitigated, for example, by better procedure and user interface design, alarm management, and team training. There is some evidence that control room operators have problems in diagnosing complicated events and multiple simultaneous events in process industry. Because of internal and external constraints and limitations, they tend to use short-cuts and rules of thumb in problem solving and decision making. Their troubleshooting performance could be improved either by increasing troubleshooters' structured knowledge and experience or by training metacognition and reflection.

Within WP4, one of the main goals has been to quantify the stress of NPP operators during simulated incident and accident scenarios. In order to examine the association between stress and crew performance, the variability in two performance measures, the performance time and the operator instructor's evaluation of the performance, were modelled with the cardiac and physical activity of the operator crews. From the six measures of operator crew performance only the information seeking performance and the performance time were associated with the physiological measures of operator stress. The movement of the operator, as measured with the accelerometers was associated only with the information seeking aspect of performance, not with the performance time. The association between information seeking performance and stress might be explained by the larger requirement for cognitive processing at the information gathering phase of the task. These results show that psychophysiological measurements of stress and activity can provide valuable information on stress and its association with cognitive performance at work.

In WP5, emergency exercises has been studied from a theoretical (resilience), regulatory (e.g., YVL guides) and practical points of view, as realised in the Finnish operating nuclear power plants. Several development needs were identified. The main issue was the lack of clearly defined objectives for the exercises. With such objectives, exercises could be planned and evaluated in a systematic way. In this way, the participating personnel would know what they are expected to do and eventually, the performance in exercises and consequently in a potential emergency situation as well would improve. It has also been found that exercise participants easily lack true feedback to learn, which is related to the evaluation of emergency exercises. With no systematic evaluation, including feedback about the appropriateness of each action performed in the emergency exercise, it is hard to develop the preparedness in personal level. The building of relevant procedures and workflows partly compensate that, but there is always the share of human factor left, in the form of error or success. False conceptions and biases in performance are worth correcting, and the new ideas and especially successful performance are the cornerstones for better performance. This is important in the personal and also in the organisational level. A step-by-step guide is provided about systematic debriefing related to an emergency exercise, serving as a means to providing feedback and gathering lessons learned also from the individual employee's point of view (Figure 2.1).

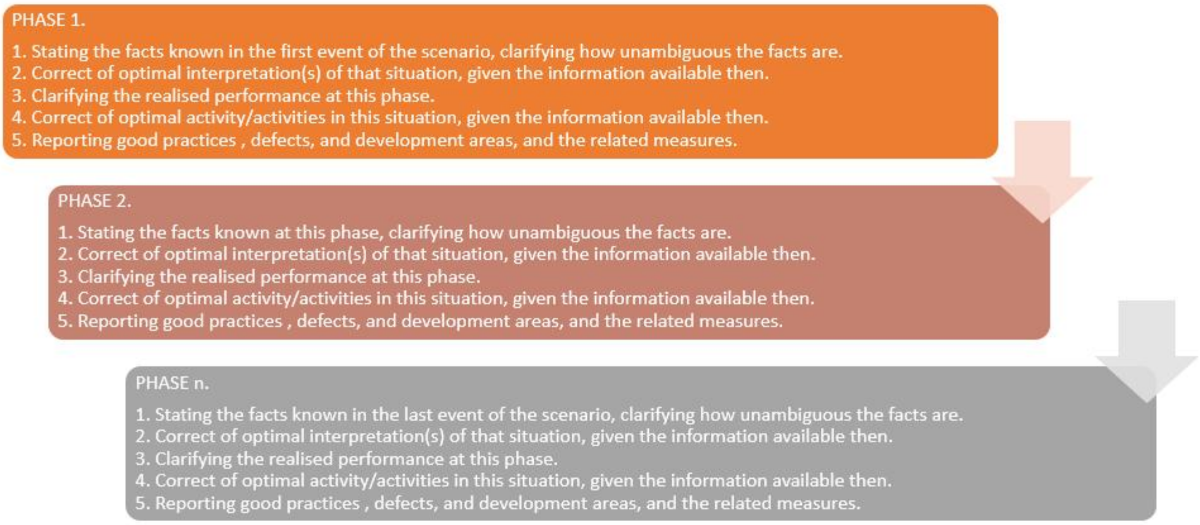


Figure 2.1. The proceeding of systematic debriefing after an emergency exercise. Each point is repeated related to each new phase in the scenario until all phases are discussed.

Within WP6, the HF-tool has been modified, trained, tested, and implemented to include basic and refresher training, detailed tool for gathering investigation data from HF perspective (HF-fan), and designing combination of Accimap and HF-tool (HF-map) for OE analysis. The HF tool was used by safety experts of two NPPs, to recognize and analyse both positive (successes) and negative (risks, errors) contributing factors to OEs. Based on the HF tool testing, the HF tool was regarded as clear and easy-to-use, and it was considered a useful tool especially at OE analysis, reporting and training as well as to self-evaluation and for monitoring safety trends. It was found to offer a more accurate picture of the analysed OEs and HFs affecting OEs including the success factors, than current OE analysis methods. The future implementation of the HF-tool would demand more systematic work with evaluating the quality of analysis and corrective actions based on findings with HF focused OE analysis, conducted by NPPs. Furthermore, collecting more accurate OE data would be a prerequisite for proper analysis of the real outcome. This would demand more HF competence and awareness not only by safety experts in NPPs, but also by the authority, top and middle management and supervisors in NPPs, operative personnel, to renew practices in safety management. For this kind of future work, we defined strategic guideline for HF implementation, reported as a part of CORE final report.

Deliverables in 2018

- A guideline leaflet entitled “Operatiivisen toimintavalmiuden kehittäminen ydinvoima-alalla” summarizing the findings of the project.

- A scientific publication presenting a socio-technical success analysis framework which was developed based on several event investigation methodologies.

- A booklet “Learning from successful operations in nuclear power plants – a guideline”.

- A journal article manuscript draft discussing the development of nuclear OE from the perspective of learning from successes.

- A journal article manuscript summarizing the past three years of studies in CORE WP2 as well as the parallel study on robotic surgery.

- A final report providing a summary and a theoretical synthesis of the work done in WP3 on multitasking and interruption management, and cognitive heuristics in decision making and troubleshooting.
- Guidance on troubleshooting and cognitive debiasing.
- A conference article on descriptive modelling of team troubleshooting in nuclear domain.
- A conference article on cognitive heuristics and biases in process control and maintenance work.
- A research article on the association of stress with cognitive performance at work.
- A final report, summarizing the main findings of WP4.
- A conference abstract on hierarchical Gaussian modelling of instantaneous heart rate distributions.
- A final report, providing a description of the present state of emergency exercises, from a theoretical, regulatory and practical points of view.
- Guidance for emergency exercise practice.
- An article draft regarding learning from operating experiences in nuclear power plants.
- A scientific manuscript on the utilization of HF perspective to renew safety management in nuclear industry.
- An invited semi-plenary talk included HUMTOOL findings, among other examples of implementing HF in safety critical fields during the last 17 years.
- Tutorial on human factors at safety critical fields.

2.1.2 EXWE - Extreme weather and nuclear power plants

The general objective of EXWE (2015–2018) was to estimate, as reliably and accurately as possible, probabilities of occurrence of extreme geophysical events that affect the design principles of nuclear power plants (NPPs) and may pose external threats to the plants. Three themes were covered in 2018: 1) extreme weather, 2) extreme sea level events, and 3) atmospheric dispersion tool. Like the first two topics, also the third theme dealt with the environment of a nuclear facility, but now impacts of the plant on the environment are considered rather than vice versa.

The focus was on low-probability events, including such very rare incidents that have not been recorded during the past 100 years of observations. In addition to that, a challenge arises from the fact that the frequency of exceptional external conditions around the NPP sites is subject to global climate change. Therefore, a hazard curve evaluated from time series of past measurements needs to be regularly updated.

In EXWE, various observations, modelling and machine learning approaches were utilized. A recent paper gave an overview on meteorological and marine studies in EXWE since 2007 to support nuclear power plant safety in Finland (Jylhä et al., 2018). Here, the main results achieved in 2018 are summarized.

Specific goals in 2018

WP1 of EXWE focused on extreme weather events. In 2018, our objective was to increase the solidity of estimates of probabilities of extreme warm-season convective weather phenomena, intense sea-effect snowfall, and freezing rain combined with strong wind by deepening our understanding of the occurrence of these events and modelling aspects of them.

1.1) Warm-season convective weather phenomena. Extreme convective weather (hereafter called ECW) is associated with thunderstorms and materializes as heavy rain, large hail, intense lightning, downbursts and/or tornadoes. The specific goals in 2018 were i) to evaluate the use of machine learning methods together with data about atmospheric conditions in order to predict the initiation of ECW; ii) to assess climate change impacts on ECW climatology from 1979 to present day; and iii) to assess probabilities of high lightning peak currents. The atmospheric conditions were taken from the ERA5 reanalysis dataset produced by the European Centre for Medium-Range Weather Forecasts (ECMWF). Based on validations against lightning observations, a predictor of thunderstorms (CAPE, convective available potential energy), being derived from ERA5, was found to capture the presence of deep, moist convection very well in Northern Europe. The time series of mean summertime CAPE over Finland from 1979 to 2018 indicated a significant positive trend (Figure 2.2), suggesting increases in flash density and thunderstorm days and/or in an increase in the intensity of the storms. According to our estimates, the upper limit of the natural lightning peak current lies close to 500 kA for negative strokes and 600 kA for positive strokes.

1.2) Intense coastal snowfall. While summertime ECW is more likely to occur, wintertime sea-effect ECW occasionally develops over ice-free sea areas. Depending on the mean wind direction, excess coastal snowfall may occur, as happened in Merikarvia on 6 January 2016. The specific goals in 2018 were to i) simulate and analyse four past coastal snowfall cases with the high-resolution convection-permitting numerical weather prediction model HARMONIE, using weather radar assimilation and to ii) to use daily observational gridded snow data as a reference data. The aim of the four case studies were to determine criteria triggering the sea-effect snowbands in Finland. Compared to criteria introduced in Sweden by Jeworrec et al. (2017), the wind speed could be slightly weaker and the wind direction enabling snowbands to hit the coast can vary much more in Finland. In addition, iii) a further, international study was made about the Merikarvia case (Olsson et al., 2018).

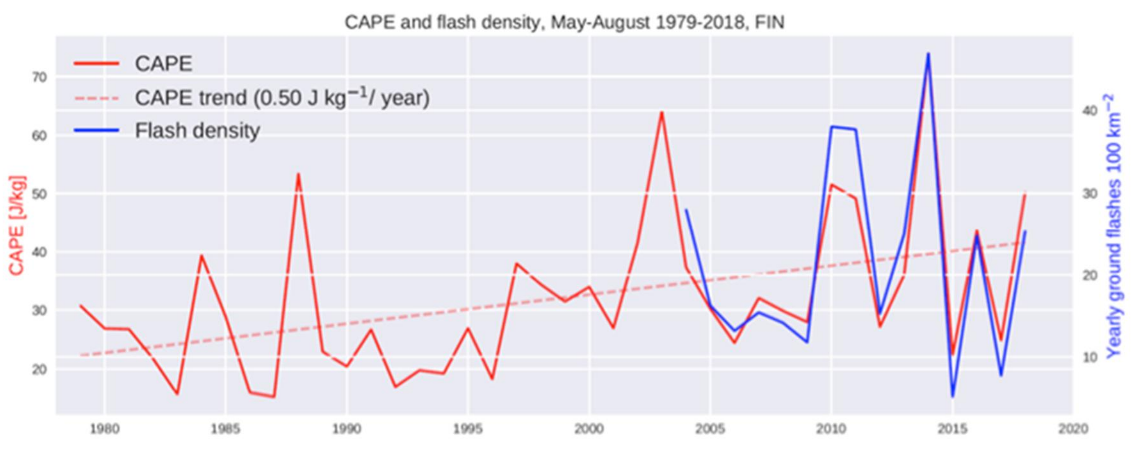


Figure 2.2. *The time series of mean summertime CAPE (solid red line) and its linear trend (dotted red) over Finland from 1979 to 2018, derived from ERA5. Also shown is the flash density for ground flashes from 2004 onwards. Since then the lightning detection network has differentiated between ground and cloud flashes and the detection efficiency has been high and relatively homogenous over Finland*.

1.3) Joint events of freezing rain and strong winds. The specific goal in 2018 was to produce statistical information about prevailing wind conditions during freezing rain (FZRA) events using the previously-developed detection algorithm (Kämäräinen et al., 2017), ERA-Interim reanalysis data and regional climate model (RCM) simulations. Gale winds (17–20 m/s) were found to rarely co-occur with FZRA, but precipitation during FZRA appeared to be somewhat more intense than in non-FZRA precipitation. In future, precipitation of non-FZRA is expected to increases somewhat, but it is unclear if there is more increase in precipitation amounts during FZRA. No apparent changes in the wind distribution were projected either in FZRA or non-FZRA cases.

WP2 of EXWE focused on extreme sea level and wave events. In 2018, extreme sea levels in the Baltic Sea were simulated and scenarios of mean sea level change on the Finnish coast were updated.

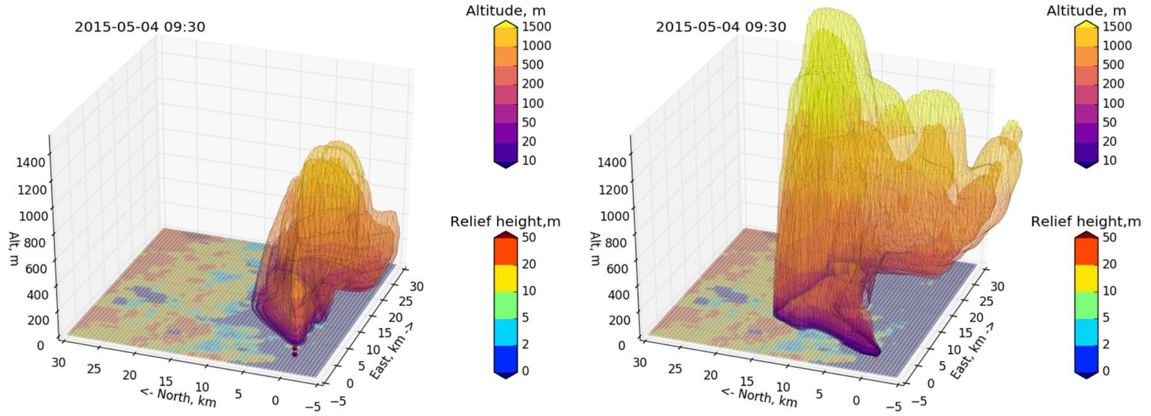
2.1) Simulating extreme sea levels in the Baltic Sea. Experiments by a numerical sea level model enable studies about extremely rare but physically plausible sea level events. In 2017, a simulation of sea level in the Baltic Sea in 1900–2010 was performed, taking into account both internal variations within the Baltic Sea basin and water volume variations due to water exchange in the Danish Straits. Simulated sea levels agreed well with tide gauge observations from Finland and Sweden, but highest sea level extremes were underestimated because of the insufficient resolution of the atmospheric forcing data from the ERA-20C reanalysis. In 2018, the simulations were analysed further. The simulated sea levels were divided into the storm surge component and the water balance component. Despite the positive trends in total sea level, the study found positive trends on the Finnish coast and negative trends in Sweden for the storm surge component. Further studies utilizing different atmospheric reanalyses for the sea level simulation are planned to be carried out.

2.2) Updated scenarios of mean sea level change on the Finnish coast. The specific goal in 2018 was to improve previous estimates of future changes in mean sea level for i) each Finnish tide gauge and for ii) each NPP site, by taking into account most recent results regarding global sea level rise and its regional deviations. Probability distributions of mean sea level change along the Finnish coast under three Representative Concentration Pathway (RCP) scenarios were combined with results from a semi-empirical land uplift model and projected changes in wind climate. The results for the NPP sites are summarized in Table 2.2.

Table 2.*2. Projected mean sea level change in 2000–2100 at the Finnish NPP sites for the three Representative Concentration Pathway (RCP) scenarios and different probability levels (Pellikka and Johansson, 2019).*

	Hanhikivi			**Olkiluoto**			**Loviisa**		
	RCP2.6	RCP4.5	RCP8.5	RCP2.6	RCP4.5	RCP8.5	RCP2.6	RCP4.5	RCP8.5
1%	-88	-80	-65	-67	-59	-44	-28	-19	-3
5%	-81	-71	-54	-61	-51	-33	-21	-10	8
50%	-61	-47	-22	-41	-26	-1	0	15	42
95%	-37	-1	59	-16	22	84	25	64	129
99%	-23	29	108	-2	54	135	39	97	181

WP3 of EXWE focused on atmospheric dispersion. The specific goal in 2018 was to make evaluations of high-resolution simulations of the FMI's dispersion model SILAM (Sofiev et al., 2006, 2015). Evaluation of local studies included a qualitative analysis of an artificial test case with an imaginary source in Loviisa. SILAM was nested with HARMONIE-AROME, a high-resolution non-hydrostatic numerical weather prediction model. For the first time, SILAM was applied at 500 m spatial resolution. The case study showed the expected features of the plume dispersion – including 3-D separation of the plume with regard to local jets, reaction to resolved convective structures, etc. (Figure 2.3). Evaluations of SILAM simulations across the whole country at three different resolutions (1, 2.5 and 10 km) indicated substantial positive impact of boosting the resolution. In particular, at 1 km, an expected sharpening of details and larger dynamic range of the near-source predictions could be seen.



a) Source height 500 m

b) Source height 10 m

Figure 2.3. *Position of the plume after 2.5 hours since the start of the release with two different heights of the source. Panel a) high source at 500 m above the ground, panel b) low source, 10 m above the ground.*

Deliverables in 2018

- A manuscript on the use of neural networks (NN) to improve reanalysis-based climatology of convective weather
- A report on the climate change impacts on severe summer convection in Finland.
- A manuscript on probabilities of high lightning peak currents
- A report on intense coastal snowfall based on case-study simulations and observational gridded data of snowfall in the NPP regions.
- A scientific paper on sea-effect snowfall case in the Baltic Sea region analysed by reanalysis, remote sensing data and convection-permitting mesoscale modelling.
- A report on joint events of freezing rain and strong winds.

- A scientific paper on recent meteorological and marine studies to support nuclear power plant safety in Finland.
- A manuscript on the results of sea level simulations.
- A manuscript on scenarios of mean sea level change on the Finnish coast.
- A short report on scenarios of mean sea level change at the NPP sites.
- A report on the assessment of the high-resolution SILAM dispersion & dose-assessment system

References

Jeworrek, J., Wu, L., Dieterich, C., Rutgersson, A., 2017: Characteristics of convective snow bands along the Swedish east coast. Earth System Dynamics, 8, 163–175.

Jylhä K, M. Kämäräinen, C. Fortelius, H. Gregow, J. Helander, O. Hyvärinen, M. Johansson, A. Karppinen, A. Korpinen, R. Kouznetsov, E. Kurzeneva, U. Leijala, A. Mäkelä, H. Pellikka, S. Saku, J. Sandberg, M. Sofiev, A. Vajda, A. Venäläinen, J. Vira, 2018: Recent meteorological and marine studies to support nuclear power plant safety in Finland. Energy, 165 (A), 1102-1118, https://doi.org/10.1016/j.energy.2018.09.033

Kämäräinen, M. et al., 2017: A method to estimate freezing rain climatology from ERA-Interim reanalysis over Europe. Natural Hazards and Earth System Sciences, 17, 243–259, https://doi.org/10.5194/nhess-2016-225.

Olsson T, Post P, Rannat K, Keernik H, Perttula T, Luomaranta A, Jylhä K, Kivi R, Voormansik T, 2018: Sea-effect snowfall case in the Baltic Sea region analysed by reanalysis, remote sensing data and convection-permitting mesoscale modelling. Geophysica, 53(1), 65-91. http://www.geophysica.fi/pdf/geophysica_2018_53_olsson.pdf .

Pellikka, H., Johansson, M., 2019: Mean sea level change at the Finnish NPP sites by 2100 under the RCP scenarios. EXWE/SAFIR2018 deliverable D2.2.2 in 2018, Finnish Meteorological Institute.

Sofiev, M., Siljamo, P., Valkama, I., Ilvonen, M. & Kukkonen, J. 2006. A dispersion modelling system SILAM and its evaluation against ETEX data. Atmospheric Environment, 40, 674–85. https://doi.org/10.1016/J.ATMOSENV.2005.09.069

Sofiev, M., Vira, J., Kouznetsov, R., Prank, M., Soares, J. & Genikhovich, E. 2015. Construction of the SILAM Eulerian atmospheric dispersion model based on the advection algorithm of Michael Galperin, Geosci.Model Developm. 8, 3497-3522, doi:10.5194/gmd-8-3497-2015.

2.1.3 MAPS - Management principles and safety culture in complex projects

The *ultimate goal of MAPS project* (2015-2018) is to enhance nuclear safety by advancing knowledge on supporting high quality execution of complex nuclear industry projects, including modernizations and new builds. MAPS is an *interdisciplinary* project, which brings together expertise in safety culture and organizational factors, governance of complex projects, project alliancing/collaborative project arrangements, societal research on safety regimes and system dynamics modelling.

The specific *objectives* of MAPS project are as follows:

1) To identify the generic safety principles of managing complex projects in the nuclear industry;

2) To clarify the cultural phenomena in major projects and the influence of time, scale, governance models, and the diversity of the involved actors on safety culture, and thus on safety;
3) To facilitate management and safety culture of ongoing and planned major projects by providing practical tools and guidance on e.g. facilitating communication, organising decision-making in unexpected situations, encouraging openness, and distributing knowledge and lessons learned.

Specific goals in 2018

In 2018 we aimed at summarizing results from the case studies in complex nuclear industry projects to provide insight on how project organisations are handling critical incidents and describing the process of sensemaking. Based on the main case study, we submitted a journal paper to the International Journal of Managing Projects in Business.

We aimed at practical tools and guidance development: we developed a first version of a *self-assessment tool* for *governance for safety in complex project networks in nuclear industry*. The tool is based on a conceptual project governance framework, developed in MAPS. The objective of the tool is to provide means and vocabulary for joint analysis and improvement of management practices in project network to meet project performance targets, and to focus attention specifically to issues that are important for ensuring that project safety goals are met. All organizations in the network, their relationships and actions influence the achievement of safety goals. Management practices are context dependent and analysis/improvement always has to take into account the specific project context. The tool was discussed at a workshop with representatives of the nuclear industry. The aim was to enhance understanding of the theoretical foundations of the tool and to discuss its potential practical applications (piloting) in the industry.

In terms of scientific publishing, our objective was to contribute to a book *Safety Science Research: Evolutions, Challenges and New Research Directions*, edited by Jean-Christophe Le Coze (INERIS), to be published in 2019 by Routledge. We wrote a book chapter on governance for safety in inter-organizational project networks. The chapter positions the MAPS project governance framework within safety-critical project context.

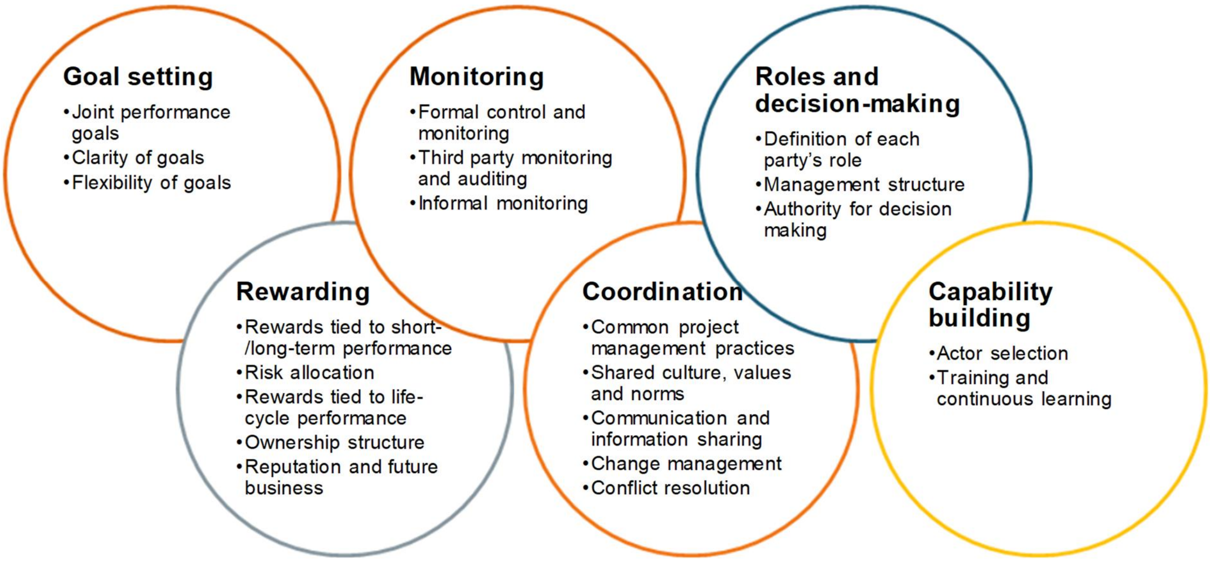


Figure 2.4. Key elements and mechanisms of governance in inter-organizational project networks (Gotcheva et al., in press; based on Kujala et al., 2016).

Regarding application of system dynamics modelling in complex projects, research work included further simulations, running and validating the model, and focused on development and validation of a system dynamics modelling training tool for project management. The tool consists of a system dynamics simulation model and an interactive user interface. We designed the use cases and scenarios for the training simulator, developed an interactive user interface (a web-based tool), gathered data on model parameter values and functions, and piloted the training with representatives of power companies and the regulator during a workshop, which also provided valuable feedback for the tool development.

Integration and dissemination of results aimed at continuing the collaboration with FIOH (HUMTOOL task in CORE project) to support facilitation of system-wide integration and co-operation of actors in the nuclear industry. We organized a joint industry seminar and wrote a joint article on the topic. Dissemination activities included participation in an IAEA meeting to finalize the SCCIP, participation in an international scientific conference (PSAM14, USA), and holding MAPS dissemination seminar to present and discuss main results with the nuclear industry community.

MAPS project was also presented as a book chapter in a new VTT edited book *Impacts from VTT Research on Nuclear Safety and Radioactive Waste Management,* Holt, E., Kinnunen, P., Sevelev, D. (Eds.) VTT Technical Research Centre of Finland, Espoo, Finland, 120 p. ISBN 978-951-38-8705-6.

Deliverables in 2018

- First version of practical self-assessment tool to improve governance and safety in complex inter-organizational networks in the nuclear industry was finalized.
- Workshop with practitioners to present and discuss about the self-assessment tool was held: the aim was to enhance understanding of the theoretical foundations of the tool and to discuss its potential practical applications (piloting) in the industry.

- A book chapter was submitted: Gotcheva, N., Aaltonen, K. and Kujala, J. (in press). Governance for safety in inter-organizational project networks*, Safety Science Research: Evolutions, Challenges and New Research Directions,* Publisher is Routledge, Editor Jean-Christophe Le Coze (INERIS). The chapter contributes to contemporary conversation in safety science by connecting discourses of safety and project governance to enhance safety in temporary and dynamic project environments.
- A system dynamics tool for project management training was developed jointly by VTT and University of Oulu. The tool provides support to decision-makers in complex nuclear industry projects based on system understanding and facilitating a holistic perspective by visualizing patterns of dynamics interrelations in the project context. This tool uses web user interface and aims at improving managers and other experts' capabilities for anticipating and acting upon unintended effects in complex projects.
- The system dynamics interactive tool for project management training was piloted with the power companies/regulator at a workshop with representatives of TVO, Fortum, Fennovoima and STUK.
- N. Gotcheva participated in an IAEA meeting related to safety culture assessment to finalize the Safety Culture Continuous Improvement Process (SCCIP) Training Material, Vienna, Austria, 9-11 April.
- Joint HUMTOOL-MAPS seminar ('system-workshop') was held on 8.11.2018 in FIOH, with 11 participants from TVO, Fortum and STUK. Collected data was utilized in a scientific manuscript. The seminar focused on discussing applicability of the HF tool to support facilitation of co-operation and system-wide integration of actors in the nuclear industry.
- Joint scientific manuscript 'Utilizing HF perspective to renew safety management in nuclear industry: design science perspective' was jointly written by HUMTOOL task in CORE project and MAPS.
- K. Viitanen (VTT) participated in a scientific conference Probabilistic Safety Assessment & Management conference (PSAM14) in September in the USA and gave a presentation on "Mapping methodical change in safety culture". The paper describes twelve principles of safety culture change, which essentially summarize good practices of safety culture change. Examples of failed safety culture change initiatives were analyzed from the perspective of these principles to provide a tentative proof-of-concept of their usefulness.
- MAPS dissemination seminar was held on 12.10.2018 at VTT with 11 representatives of the Finnish nuclear industry and the regulator STUK. The seminar focused on presentation of key results from the MAPS project and discussion with practitioners.

2.1.4 PRAMEA - Probabilistic risk assessment method development and applications

The general objective of the PRAMEA project is to develop methods and tools for probabilistic risk analysis (PRA) of digital systems and to utilize them in practical case studies. The project covered most of the topics relevant to the PRA of nuclear power plants. PRAMEA has conducted a human error analysis by interviewing operators of an NPP, focusing on the management of a primary to secondary leak in a hybrid control room. It has participated in the preparation of an IAEA safety report for HRA. It has developed methods and conducted pilots for site PRA in Nordic co-operation project SITRON. Further, it has demonstrated different options to model emergency core cooling system recovery time and the effects of different timings in level 2 PRA, and conducted advanced uncertainty analysis. It has developed a new time-dependent calculation feature in the FinPSA PRA code, and improved its risk integrator.

In level 3 PRA, it has conducted a survey among the Finnish nuclear stakeholders on the incorporation of seasonal and contextual factors in level 3 analyses and consequence assessment. Finally, it has developed a method to select mitigation strategies for protecting electric power systems from cyber threats.

Specific goals in 2018

The objectives of the task on HRA of digitalized control rooms were to define the characteristics and trends for computerized HSI, including computerized procedures, in modernized and new control rooms and conduct initial work on task analysis or Performance Shaping Factors.
To participate in the international expert team to prepare a guide on human reliability analysis. The work is led by the International Atomic Energy Agency.

To develop a Nordic approach for a nuclear power plant site risk analysis, driven by performance of pilot studies. This work includes a study on the role of technical support centre (more generally emergency response organisation) in multi-unit scenarios.

The objective on level 2 integrated deterministic and probabilistic safety analysis (IDPSA) task was to continue the development of previously developed fictive boiling water reactor (BWR) model and study handling of different types of uncertainties in FinPSA level 2.

Level 2 method support objectives were related to FinPSA code. They included computation of total release results covering all level 2 accident sequences. Total results can be extended to cover similar results as calculated for each release category (e.g. distributions, statistical parameters and correlation results for each collected variable).
An objective was to develop time-dependent PRA computation in FinPSA, e.g. taking into account planned maintenance operations.

In level 3 PRA, the objective was to conduct a survey among the Finnish nuclear stakeholders to find out about the significance, current status of incorporation, and future prospects of the incorporation of seasonal and contextual factors in analyses in the Finnish context.

In the risk analysis of organizations and operations, the objective was extend the methodology, developed in previous years within PRAMEA, to support the selection of cost-efficient portfolios of structural safety measures for dynamic systems.

Deliverables in 2018

- A review paper has been submitted to *Reliability Engineering and Safety Science, Special Issue on Foundations and novel domains for Human Reliability Analysis*: Markus Porthin, Terhi Kling, Marja Liinasuo. Effects of Digitalization of Nuclear Power Plant Control Rooms on Human Reliability Analysis – a Review (currently in review process).

- A case study was carried out to identify possible human errors related to accident management. The study focused on human performance with the procedures for PRISE in a hybrid control room. Interviewing of operators was used as the research method. .

- Participation in an IAEA expert group to develop a Safety Report on Human Reliability Assessment for Nuclear Installations. Draft report prepared during 2018 and one meeting organised in Vienna. The report will go to IAEA's publishing process in 2019.

- A method for site PRA dedicated to Nordic conditions where NPP units have comprehensive single-unit PRAs available. The approach has been presented also in several international conferences (ESREL2018, PSAM14, OECD NEA workshop July 2018, IAEA coordinated research project meeting June 2018).

- We have prepared guidelines for site PRA model management and detailed requirements for site PRA data base.

- Two pilot studies on site PRA (Forsmark 1&2 units and Ringhals 3&4 units). Pilot studies demonstrate the feasibility of the proposed approach to site PRA.

- Survey on the role of the role of technical support centre (more generally emergency response organisation) in multi-unit scenarios. The results show the differences and similarities in the implementation of emergency response organisation at sites in Finland and Sweden and the study also discusses the assessment of technical support centre in the context of PRA. For time being, technical support centre is only considered a limited way in PRAs.

- We have used simple emergency core cooling system recovery case to demonstrate different options to model the recovery time and the effects of different timings. The case study demonstrates the need to separate epistemic and aleatoric uncertainties in dynamic accident modelling in order to analyze uncertainties in a consistent manner. We have also developed a high pressure melting containment event tree of the previously developed simplified BWR model further.

- For FinPSA, the risk integrator has been updated and tested to compute total results as calculated for each release category. Additionally, containment event tree (CET) contributions to frequencies of release categories, results for category (and Boolean) based variables, and CET sequence and branch function contributions to source variables and frequency of release categories (or total results) have been implemented and tested. We have developed a new time-dependent calculation feature in FinPSA. The user can specify events, such as maintenance activities or tests, parameter changes and configuration changes to a time line in a task file, and FinPSA automatically calculates and draws the time-dependent risk curve based on that information, existing minimal cut sets and data.

- We conducted a survey among the Finnish nuclear stakeholders (STUK, Fennovoima, Fortum, TVO) on the significance of seasonal and contextual factors (SCF) to the risks of Finnish nuclear power plant sites, on the current incorporation of these in level 3 PRA and related analyses, and on future prospects and research needs related to PRA involving SCFs. Level 3 analyses have not been carried out in Finland thus far, but seasonal factors have been taken into account in consequence analyses. Some SCFs (e.g. snow cover) have been found to affect accident consequences significantly. The results were documented in a research report.

- In 2017, we developed a methodology to support the selection of cost-efficient portfolios of structural safety measures for dynamic systems. In particular, we extended the methodology to time-dependent accident scenarios by explicitly encoding the dynamic behaviour of engineering systems. In late 2018, the manuscript has been conditionally accepted by *Reliability Engineering and System Safety*.

- In 2018, we developed a methodology to select mitigation strategies for protecting electric power systems from cyber threats. This paper has been developed in collaboration with the International Institute for Applied Systems Analysis (IIASA, Austria). The manuscript has been submitted to *IEEE Transactions on Dependable and Secure Computing*.

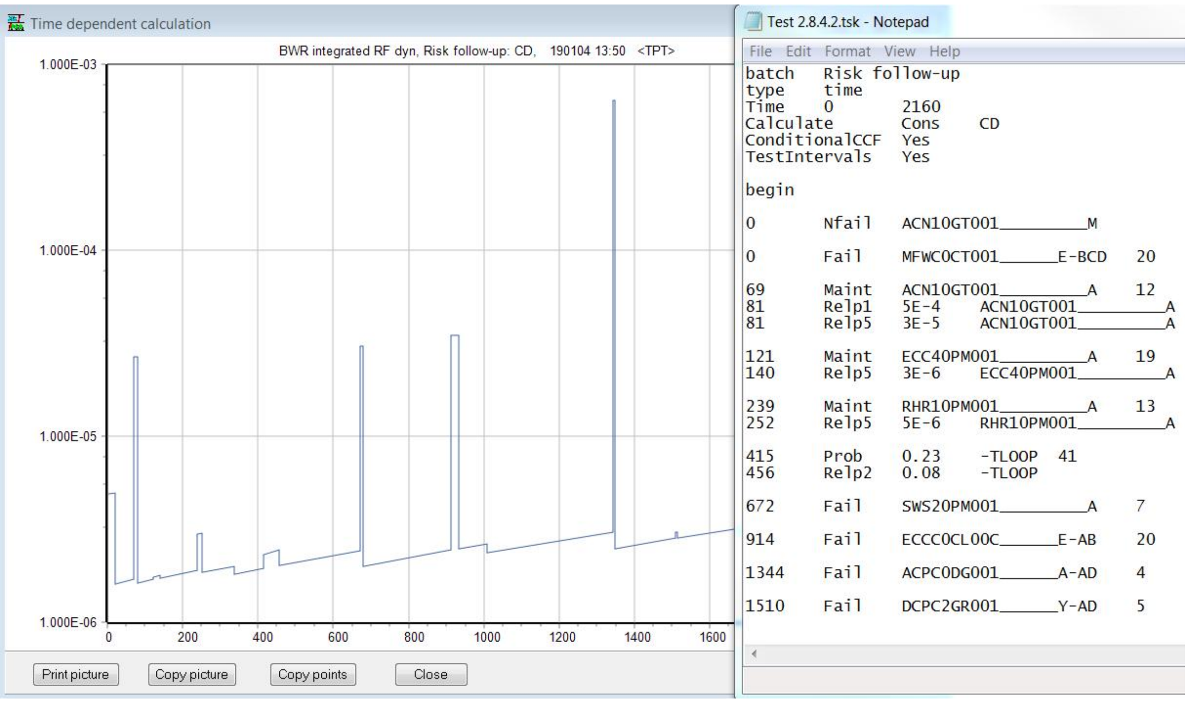


Figure 2.5. Time-dependent task file and the resulting risk curve.

2.1.5 SAUNA - Integrated safety assessment and justification of nuclear power plant automation

The general objective of SAUNA (2015-2018) has been to develop an integrated framework for safety assessment and transparent safety demonstration of nuclear power plant instrumentation and control (I&C) systems. Traditionally, safety assessment tends to focus on technical issues, single faults or limited combinations, deterministic analysis and probabilistic risk assessment (PRA), and a document-based approach (SARs). In SAUNA, the research theme was overall plant safety. A key challenge was therefore to consider (1) all types of hazards, including rare and extreme conditions, (2) all disciplines and types of system elements (technical, human, environmental…), and (3) all life-cycle phases and activities.

Through a multidisciplinary research strategy, SAUNA looked at plant operations in the context of the whole plant and investment project, while keeping the focus on I&C systems. The different work packages (1) built a shared understanding of the underlying challenges, concepts, and Systems Engineering principles, (2) developed dedicated methods and tools for assessing the safety of systems and their development processes – particularly focusing on the efficient integration of different approaches – and (3) aimed to tie the results together into an integrated, structured, model-based approach to safety demonstration and licensing.

Specific goals in 2018

WP1 of SAUNA focused on clarifying the terminology and Systems Engineering principles in order to provide a common basis for research activities on NPP safety. In 2018, the focus was on further developing model-based safety assessment methodologies for Defence-in-Depth, also touching on security issues.

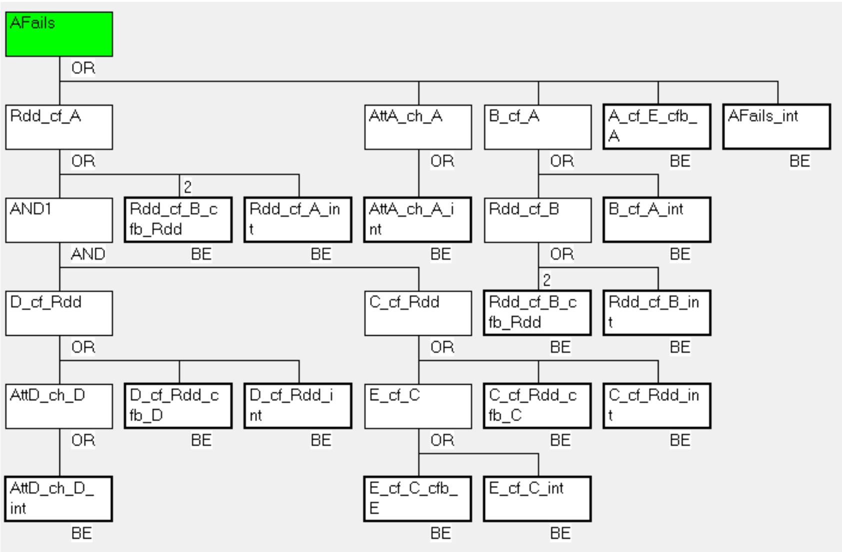


Figure 2.6.. An example of a hybrid threat and fault tree of a system with five components (A to E), four attackers and one redundancy Rdd between components C and D. The tree shows failure propagation paths triggered by either faults (causes failure relationships "_cf_") or vulnerabilities (causes harm relationships "_ch_"). The tree captures failure back propagation, i.e. a component C1 without failing causes another component C2 to fail which in turn causes a failure in C1 ("C1_cf_C2_cfb_C1" relationship). An example can be an overheating pump that damages its actuator electronics that in turn damage the pump.

A multidisciplinary model of the plant contains a significant amount of potential propagation paths for failures initiated by faults or security threats. In 2017, a method was developed for compiling a High Level Interdisciplinary model (HLIM) of a complex system, based on fault trees. In 2018, the dependency model behind the HLIM was extended to include security threats as basic events, leading to hybrid threat/fault trees (Figure 2.6). The method can be used for early hybrid assessment of safety and security. A prototype software implementation was demonstrated using a spent fuel pool cooling system as a case study.

Modelling of digital I&C systems in probabilistic risk assessment (PRA) remains a challenge. In 2018, PRA was applied to risk-informed safety classification of structures, systems and components related to electric and I&C systems. At the plant level, the safety importance (and therefore classification) of functions is usually clear, but on the level of smaller components, classification is not straightforward. E.g., downgrading is possible under certain conditions. A risk-informed classification approach was outlined, using an emergency diesel generator system as an example.

A workshop on risk-informed decision making in safety engineering was organized in November, summarising the results of SAUNA, and discussing future research needs. The workshop was attended by 18 participants representing all major stakeholders.

The DIGREL reference PRA model developed earlier in cooperation with Nordic partners is now being used as basis for a benchmark study in a project called DIGMAP initiated by the Working Group on Risk Assessment (WGRISK) of OECD Nuclear Energy Agency. In 2018, the goal was to develop a tentative version of a PRA model to be compared with DIGMAP partners in 2019.

On the topic of co-use of PRA and deterministic methods, an approach was developed to specify the locations of potential hardware failures in the context of application software model checking. By using the minimal cut sets of PRA to identify the hardware components whose failure is most critical, and then only allowing those components to fail in model checking, verification of the SW+HW model was made computationally feasible.

WP2 of SAUNA looked at the assessment methods and tools for specific technical, human, and systemic safety issues. In 2018, the focus was on broadening the scope and increasing the reliability of model checking, collecting the knowhow on the Nuclear SPICE process assessment method into a concise handbook, and developing practical methods on evaluating and validating control room design.

Model checking is a powerful formal verification method that has for the last ten years been successfully used in the Finnish nuclear industry. One challenge is that I&C (software) application logic is modelled as if it operated "in a vacuum", without accounting for the limitations and weaknesses of the underlying hardware. In 2018, a method was developed that introduced hardware failure modes and communication delays into the I&C software models. While the communication delay and asynchrony issues remain challenging to analyse, the method proved successfully in verifying tolerance against single failure (including consequential failure) (see Figure 2.7).

The formalization of (natural language) functional I&C requirements into the temporal logic properties used in model checking is also very challenging. A survey of different approaches (e.g., statistical machine translation) on automatically generating the formal properties was conducted. Approaches based on theory of formal languages and grammars seem to hold potential.

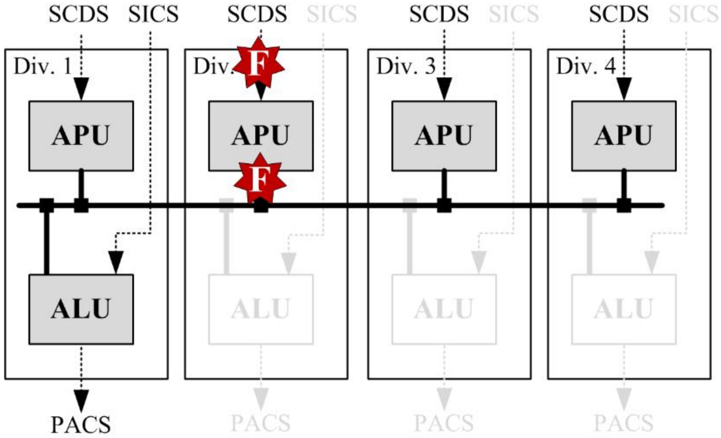


Figure 2.7. Based on the symmetry of four-channel safety I&C systems, the hardware failure model can be simplified, still enabling the verification of single fault tolerance (N+2, N+1 criteria) for software.

The Nuclear SPICE process assessment method provides a cost-efficient way to reduce risks in deliveries and collect evidence for system qualification (Figure 2.8). In 2018, a study aimed at validating the results of compliance evaluation by analysing assessment findings and their use in the domain. Finally, the Nuclear SPICE Handbook was released in a workshop in November, attended by 15 participants representing all major stakeholders. The Handbook introduces the method, defines the process assessment model, and describes the assessment process.

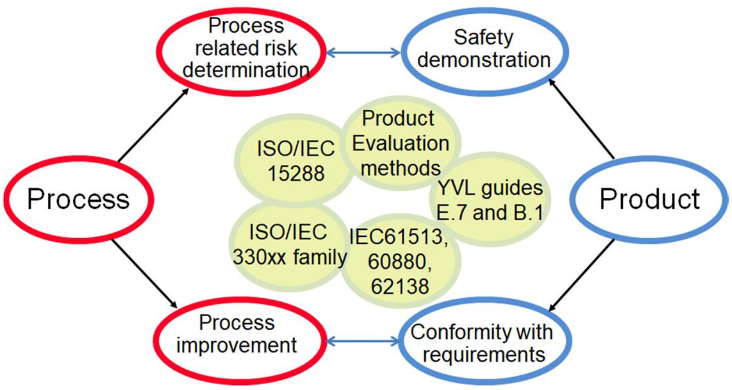


Figure 2.8. Use cases of the Nuclear SPICE process assessment

Simulators are important for training and licensing of operators, and play an important role in control room system evaluation. How the simulator resembles the control room (and the controlled process) is therefore very relevant. In 2018, a practical method for training simulator fidelity evaluation was developed, based on an expert elicitation process. The method will be applied in the feasibility study of the Fortum LOKS 2 simulator.

WP3 of SAUNA aimed to provide recommendations, insight, new viewpoints, and tools for planning, documenting and communicating the safety demonstration, and enabling the licensees to efficiently carry out the licensing process.

Model-Based Systems Engineering (MBSE) has the potential to support qualification in a cost-effective way. In 2016, a systems engineering data model was developed, and in 2017, the model was updated to encompass third party conformity assessment artefacts. In 2018, the resulting conformity assessment data model was demonstrated using the Polarion Application Lifecycle Management (PLM) tool by Siemens. PLM was used to manage the traceability and conformity claiming of a diversity requirement, using a spent fuel cooling pool system as the case study. The demonstration was successful, with the effort used for configuration deemed acceptable.
Architecture Description Languages (ADL) provide a way to analyse both hardware and software components on different levels in one model, and could also support the automatic generation of assurance case material. In 2018, the different possibilities of ADL were studied in co-operation with HRP (Halden Reactor Project). The work included hands-on architecture modelling with Architecture Analysis & Description Language (AADL) and SysML, with a case study based on the APR1400 reactor protection system. Results argue that use of ADLs offer benefits over natural languages in clarity, ambiguity and traceability. SysML was found more suitable for conceptualisation and functional analysis, while AADL has more potential in detailed, component-level, non-functional analyses.

Deliverables in 2018

- A conference paper in RAMS 2019 about safety and security assessment based on interdisciplinary dependency models

- A conference paper in ESREL 2018 about risk-informed safety classification of electric and I&C systems

- A workshop on risk-informed decision making in safety engineering

- A research report describing a benchmark system PRA model

- A conference paper submitted to ESREL 2019 about the co-use of model checking and PSA to assess failure tolerance

- A conference paper in IEEE ICIT 2019 about the modelling of hardware failures in software model checking

- A conference paper submitted to IEEE INDIN 2019 about ways of formalizing natural language requirements into temporal logic

- A research report on assessing the correctness of formal models by comparing against simulation models

- A research report that is a handbook describing the use of the Nuclear SPICE process assessment method

- A conference paper in EuroSPI 2018 about validating the results of compliance evaluation by analysing assessment findings and their use
- A slide set about a practical method for control room simulator fidelity evaluation
- A conference paper in ANS NPIC & HMIT 2017 about the use of immersive 3D technology in control room validation
- A journal article in Nuclear Technology about the use of immersive 3D technology in control room validation
- A conference paper submitted to IEEE INDIN 2019 about the demonstration of a conformity assessment data model based on an industrial case example
- A conference paper in ANS NPIC & HMIT 2019 about the applicability of SysML and AADL in safety assurance of I&C systems

2.1.6 GENXFIN - Safety of new reactor technologies

The main objective of the GENXFIN project was to increase knowledge on safety issues of advanced reactor concepts and to coordinate participation in various international forums and working groups, as well as to disseminate information to Finnish stakeholders. The emphasis of the project was on SMR (Small Modular Reactor) designs that could be deployed in Finland in the near future.

In addition to the normal VYR and VTT funding, TVO and Fennovoima directly funded the project. As an in-kind contribution, Fortum shared reports of its own SMR research to the SAFIR reference group.

Specific goals in 2018

A review of international developments in sizing of emergency preparedness zones was made, with a focus on SMRs. Since the fission product inventory in SMRs is smaller than in large reactors, also the environmental consequences of a severe accident are expected to be smaller. In addition, the lower decay heat and the use of passive safety systems may reduce both the probability of a severe accident and the probability of a containment failure. Thus, SMRs could be located closer to cities, which would allow them to be used for district heating. On the other hand, if several SMRs are located on the same site, the possibility of a multi-unit accident due to a common cause needs to be taken into account.

An ideal procedure to determine the size of an emergency preparedness zone would involve a full-scope level 3 PRA. However, this would require very large resources. Still, the definition of an acceptable risk would remain a political question: “How safe is safe enough?” A practical starting point is proposed: Select some SMR types that could be viable alternatives for Finland. Study their fission product inventories and accident mitigation systems on the basis of public sources. Then, choose some possible sites for district heat generation and study their environment in detail (population distribution, schools, hospitals, etc.) Calculate doses to the population using some justified fractions of the fission product inventory released to the atmosphere in accidents. Then check if STUK dose limits would be exceeded and how the protective measures could be carried out in practice.

Many SMR designs include passive safety systems for removing decay heat from the reactor. The SMART design has the Passive Residual Heat Removal System, and the NuScale design has the passive Decay Heat Removal System. Both systems have a vertical heat exchanger immersed in a water pool. Steam, coming from the steam line, flows through the heat exchanger tubes and condenses there. The condensed water flows back to the feedwater pipe. Passive condensers with vertical heat exchange tubes are also used in the AES-2006 steam generator passive heat removal system.

In the GENXFIN project, PANDA Isolation Condenser experiments were modeled with the MELCOR code. The work was a continuation of earlier research, in which experiments on steam condensation in vertical tubes were calculated with MELCOR. In the earlier work, it was found that in some cases MELCOR significantly underestimates the condensation rate in vertical tubes, when using default heat transfer parameters, both with pure steam and with steam–air mixtures. The problem was fixed by changing the condensate film Reynolds number limits from the default values to literature values, using MELCOR's sensitivity coefficients. The Reynolds number limits are used for classifying the condensate film flow as laminar or turbulent. The PANDA experiments calculated in the GENXFIN project were performed at higher pressure than the earlier work, and the pure steam tests were performed so that all the steam condensed in the tubes and only water flowed out.

MELCOR nodalization of the PANDA isolation condenser is presented in Figure 2.9. The condenser was immersed in a water pool at the saturation temperature and atmospheric pressure. Steam, air and helium were fed to the upper drum through the supply line. The steam condensed in 20 vertical tubes with inner diameter of 51 mm and length about 2 m. The condensed water flowed from the lower drum to the drain line. In tests with non-condensable gases, the air, helium and remaining steam flowed through the vent line to the wetwell, which was kept at a constant pressure. In pure steam tests, the vent line was closed and the condenser settled itself to an equilibrium pressure, in which all the steam condensed.

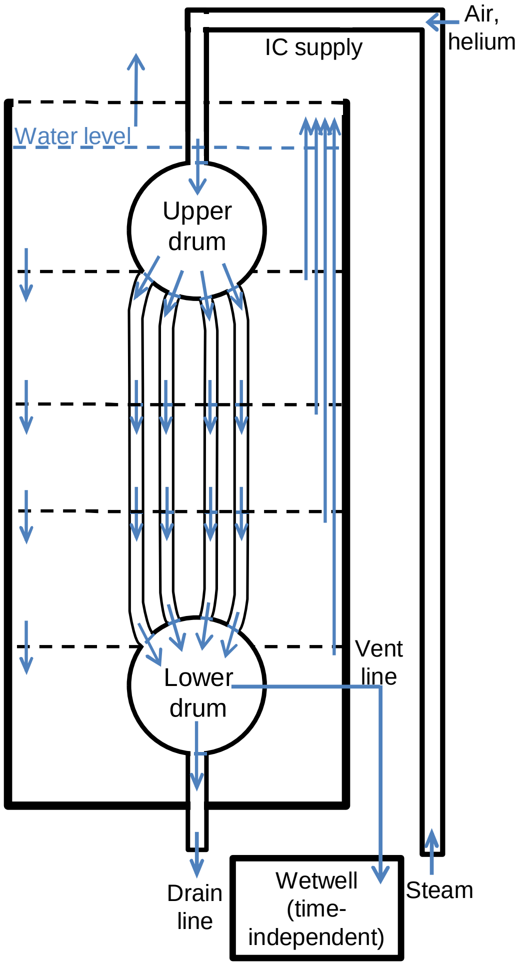


Figure 2.9. *MELCOR model of the PANDA isolation condenser. The blue arrows are flow paths.*

In the MELCOR model, the 20 heat exchange tubes were divided into four groups. Each group represented five tubes of equal length. Each tube group was divided into three control volumes (CV) of equal length in the vertical direction.

The water pool at the secondary side of the condenser was divided to five CVs in the vertical direction, in order to model the effect of the hydrostatic pressure on the saturation temperature. At first, it was attempted to model both water flow downwards and steam flow upwards with a single flow path between each pair of pool CVs on top of each other. However, the counter-current flow of water and steam through the same flow path lead to heavily oscillating flow rates and very short time steps. The problem was solved by modeling the water and steam flows with separate flow paths, with the steam paths leading from the top of each CV to the top of the whole pool. Modeling the pool as a single CV was also tested.

The calculations were made with MELCOR version 2.2 revision 11932. Four variants of the MELCOR model were used. The water pool on the secondary side was modeled either as five CVs stacked in the vertical direction, or as a single CV. Both default and modified condensate film Reynolds number limits were tested.

MELCOR underestimated the equilibrium pressure in the pure steam tests (Figure 2.10). The average underestimation was 12 % with the five CV pool model and modified Re limits. Underestimating the pressure in these tests means overestimating the condensation. As a result, the best results were obtained with the five CV pool model with the default Re limits because it gave the lowest condensation rate. This is in contrast to the earlier calculations, in which the default Re limits underestimated the condensation rate but the modified limits gave good results. The main difference is that all the steam condensed in the PANDA pure steam tests, while in the tests calculated earlier, the flow rate was so high that some of the steam did not condense.

MELCOR underestimated the condensation rate in the mixture tests (Figure 2.11). The average underestimation was 5 % with the five CV pool model and modified Re limits. The best results were obtained with the single CV pool model and modified Re limits. However, the difference between the five CV and single CV pool models was very small in the mixture tests because the main heat transfer resistance is inside the tubes, due to the non-condensable gases, and the heat transfer model on the outer surface is less important.

Using the modified condensate film Re limits can still be recommended because they give better results in all the condensation experiments calculated so far by VTT, except the PANDA pure steam tests in which the vent line was closed. Dividing the water pool into several CVs in the vertical direction may be beneficial in pure steam cases because accuracy of the heat transfer model on the outer surface of the tubes is more important than with steam – non-condensable gas mixtures. Division of the pool into several CVs requires careful definition of the flow paths between the CVs, in order to avoid oscillating steam flow that causes unacceptably short time steps and slow calculation.

The MELCOR results are compared with earlier RELAP5 and APROS calculations in figures 2 and 3. MELCOR gives about equally good results as the APROS code. The MELCOR results are better than the RELAP5 results.

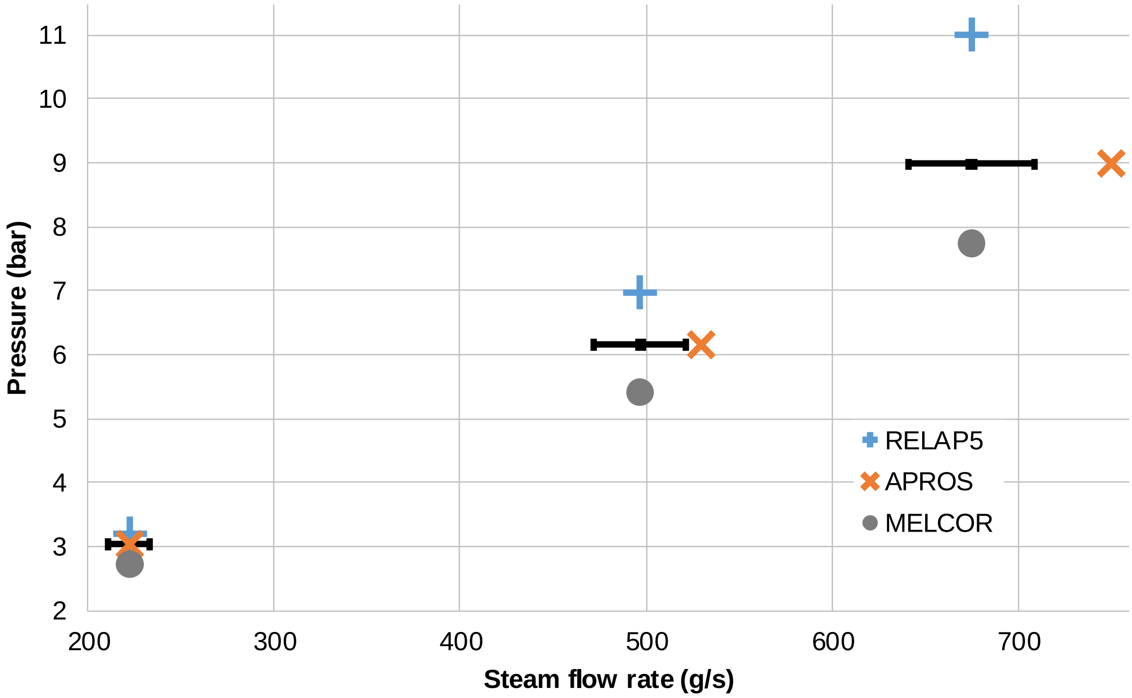


Figure 2.10. *Results of pure steam tests in the PANDA isolation condenser. The measurements are shown with the black error bars. MELCOR results from the GENXFIN project using the 5 CV pool with modified Re limits, compared with earlier RELAP5 and APROS results.*

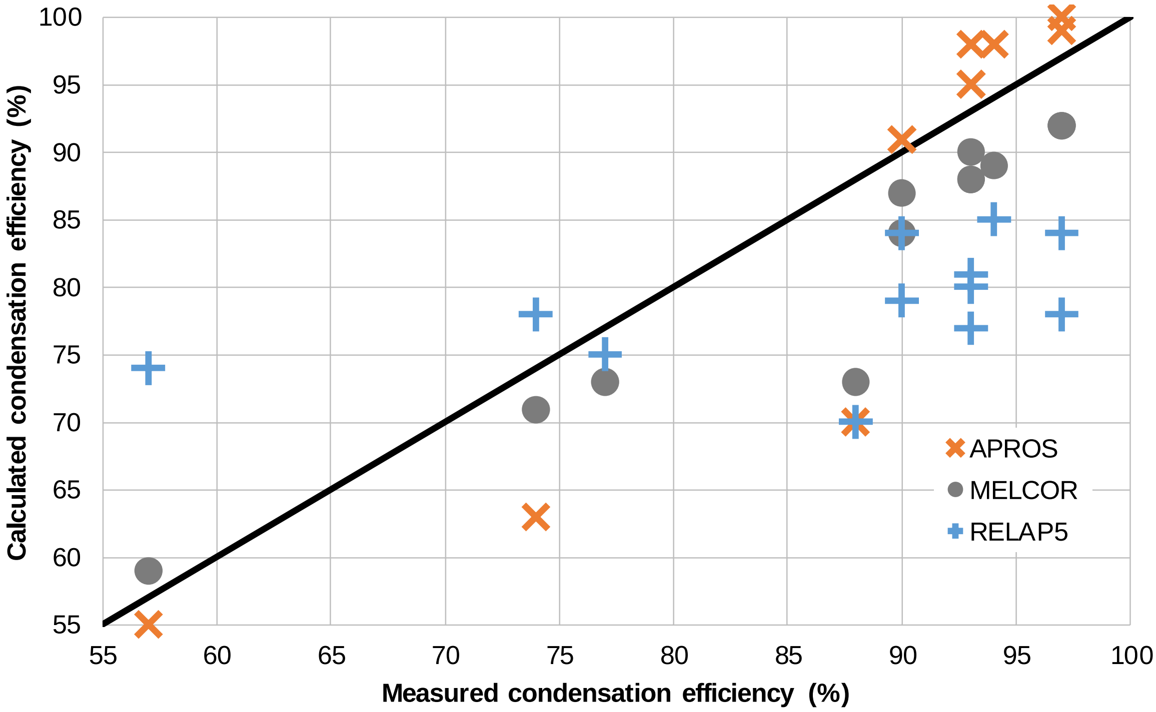


Figure 2.11. *Results of steam – non-condensable gas mixture tests in the PANDA isolation condenser. MELCOR results from the GENXFIN project using the 5 CV pool with modified Re limits, compared with earlier RELAP5 and APROS results.*

As an in-kind contribution to the GENXFIN project, Fortum shared a report of its SMR modeling with the Apros code. Two reactor types were modeled, NuScale and Yanlong. Both models are based on publicly available design data. The objectives were to test the capabilities of Apros to model helicoil steam generators and connection of an SMR to a district heating network.

In the NuScale model, two 160 MW_{th} reactors were connected to a single turbine and to a district heating network. The model demonstrated that SMR combined heat and power plants can be modeled with Apros. The simulated values are very close to NuScale design values at full power. The evaluated heat transfer rate of the helicoil steam generators is 10 % lower than in the design. The difference has to be compensated in the model either by increasing the secondary mass flow rate or the heat transfer area. Some instability was noticed in the behavior of the steam generator. If the intake pressure drops below a critical limit, in this case 1 bar below the design pressure, the steam generator start to pulsate. The cause of the phenomenon is still under investigation.

The second SMR study investigated the Chinese Yanlong 400 MW district heating reactor. The reactor is located in a 25 m deep water pool. The pool surface is at the atmospheric pressure, and the core is pressurized by the hydrostatic pressure. The model was tested at full power operation with the district heating network connection. It was concluded that the reactor concept is very robust due to its simplicity and the large amount of water.

A major part of the GENXFIN project was participation in various international forums and working groups on advanced reactor concepts, and disseminating the information to the SAFIR reference group. Meetings organized by IAEA, Generation IV International Forum, American Nuclear Society, and Canadian Nuclear Laboratories were participated. Topics of the meetings included SMRs, supercritical water reactors, district heating reactors, and accident tolerant fuels. Six travel reports were written in 2018.

In order to enhance national collaboration and information exchange on SMRs, the GENXFIN project organized a seminar in Espoo on 20 November 2018. 24 people participated the seminar. Presentations were given by VTT, STUK, Fortum, and LUT. Discussion topics included the potential of district heating with SMRs in Finland, the possibility of unmanned remote-operated reactors, siting requirements of SMRs, application of defense in depth in SMRs, and the necessity of changes to Finnish legislation and YVL guides to facilitate SMR licensing.

Deliverables in 2018

- A research report on a literature study of SMR emergency zones
- A research report describing modeling of PANDA isolation condenser experiments with MELCOR.
- A research report describing modelling of SMRs with Apros by Fortum.
- Six travel reports of international meetings organized by IAEA, Generation IV International Forum, American Nuclear Society, and Canadian Nuclear Laboratories.
- Organized a national SMR seminar.

2.1.7 ESSI - Electric systems and safety in Finnish NPP

The objective was to research 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).

Specific goals in 2018

WP1 studied different alternatives for detecting the OPC situation and their suitability for different types of OPC situations. The currently used protection methods and operating procedures were analysed and recommendations about how to improve the security of NPPs in the case of OPC were formulated.

Detection of OPCs is not straightforward, because transformers and motors re-generate voltage, and the unbalances can remain at a level, which is not detected by protection. Transformers between the faulted point and the point of interest affect significantly the observed voltages and currents. The most severe locations for OPC are the main generator bus, primary side of the unit transformer and primary side of the standby transformers. From OPC detection point of view, the most challenging OPC occurs when the NPP is supplied from the off-site grid, the main generator is disconnected i.e. transformer loading is low and the single open phase is on the primary side of the unit or standby transformer. The resulting unsymmetrical system affects the different NPP electrical system components in various ways, which is summarized in Table 2.3. The table also presents the potential NPP component protection functions that might operate in OPC situations. Several alternative methods for OPC detection exist and the selection of the most suitable methods for a particular NPP depends on the NPP characteristics. The preparedness of Finnish NPPs against OPC can be considered good and no critical safety risks were identified during this research.

Table 2.3. Effects of OPCs on NPP electrical components and the related protection of the components.

Component	**Effects**	**Protection**
Main generator	Negative sequence currents heat the rotor and can lead to damage of the generator if the unbalance situation remains for a too long time. There is also a risk of a pole slip.	· Negative sequence current relays with inverse time characteristics · Undervoltage protection · Pole slipping protection
Induction motor	If voltages are unbalanced, problems related to overheating and increasing vibrations can occur. A motor may also stall and it may not start-up with unbalanced voltages.	· Some critical motors have negative sequence current protection · Some motors have undervoltage protection · Overload protection · Some motors have temperature measurements with alarms
Power electronics (converters)	Unbalanced voltages can cause synchronization difficulties for some power electronic devices. Undervoltage and/or voltage unbalance protection can be quite sensitive and devices can disconnect easily. Possibly connects automatically back when voltage normalized.	· Differential current protection · Ground fault protection
Transformers	Not considered as the most vulnerable component during OPC. Overheating can occur in some loading conditions.	· Differential current protection · Ground fault protection

WP2 analysed over-voltages and required protection solutions in different NPP network topologies and at different voltage levels of the networks, using the simulation methods developed in 2017. A total new part of the analysis was GPR (Ground Potential Rise) in cases where lightning strikes in various grounding system parts. Analyses covered voltage stresses due to GPR in electrical, automation and control systems and possible mitigation means of the GPR overvoltages. The work involved also in modelling low voltage AC and DC systems related to over voltage protection of low voltage power electronics.

The three-dimensional finite-difference time- domain (FDTD) method has been employed to study the performance of the large-scale grounding system (LSGS) against a lightning strike with different lightning surge propagation properties and how to improve transient protection for control equipment. The study emphasized how a nearby sea influences the ground potential rise on large- scale grounding system considering soil ionization. Two case studies were conceived regarding the position of the nearby sea, where it is located on the y+ and x+ sides of the LSGS, see Figure 2.12 In addition, two striking scenarios were considered where the lightning strikes the first and the second tower. One example of simulation results is in Figure 2.13, corresponding to Figure 2.12 case.

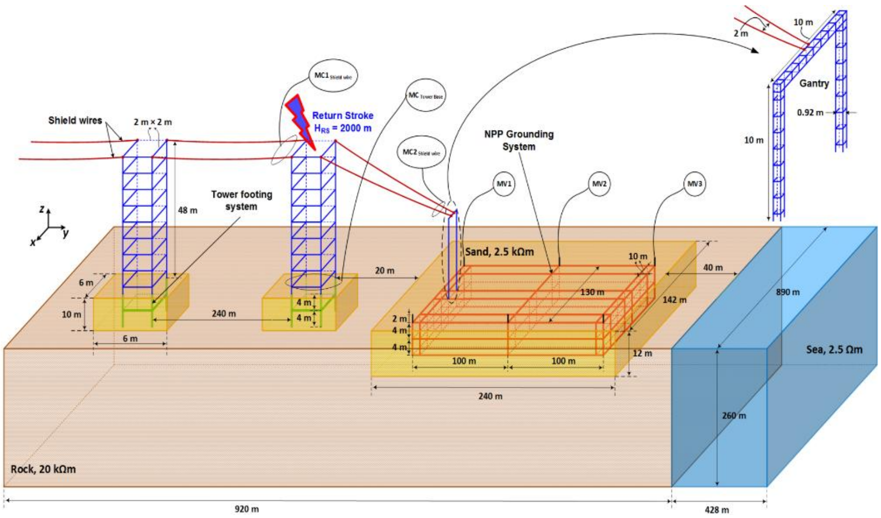
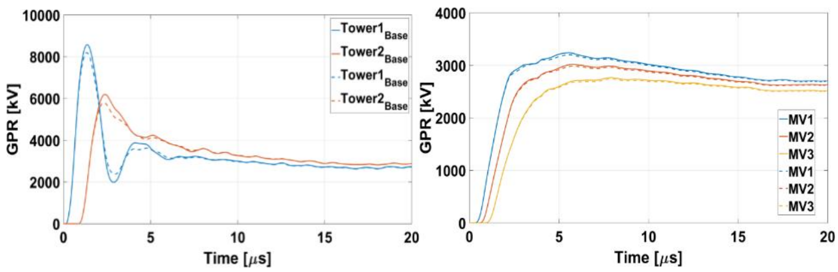


Figure 2.12. Physical grounding system model used in simulation. The figure corresponds the case where the sea is located on y+ side and the lightning strikes the first tower.



a) b)

Figure 2.13. Ground potential rise: a) in tower bases and b) in three different points of grounding system, solid line without ionization and dash line with ionization. The lightning current has a peak value of 100 kA, the rise time of strike is 4 µs and the resistivity is 20/2.5 kΩm for rock/sand.

The results showed that the magnitude of ground potential rise (GRP) at different points of LSGS is strongly influenced by sea. In addition, the extent that soil ionization affects the ground potential rise is dependent on the distance between tower and the large-scale grounding system. Soil ionization causes a decrease in the soil resisitivity and subsequently, in the computed GPR. The farther the tower is from LSGS the ionization impact on the GRP and currents waveforms becomes more significant. The potential differences on different parts of the grounding system may damage the connected wiring and equipment, depending on how grounding connections are made. For example, control devices can be grounded locally or grounded via the cable shield only since both cases can be encountered even in the same plant.

In the case of large wide area grounding system and high soil resistivity, the ground potential rise may cause high voltage stress in the connected equipment. One of the critical situations is the lightning strike to the gantry or adjacent transmission tower of a large power generating station. A large lightning current may cause huge potential differences between different grounded parts of the electrical systems. In the extreme case, these differences may cause excessive stress to the insulation of the signal cables of the automation and control systems. These voltage stresses may be mitigated by adding to the grounding system external conductive cables parallel to the protected signal cables. In addition, surge protective devices may be needed in most critical locations. Planning such a protection requires a detailed analysis of the grounding system and electrical system lay out, as well as investigation of the different routes along which the lightning current may enter the plant. It might be necessary to limit the possible lightning entering routes by using lightning rods.

The impacts of lightning strike on low voltage power electronic devices and their protection were simulated with PSCAD transient simulator. Metal oxide protector, three phase rectifier load and clamp style protection device with over-current protection and a battery were simulated. Metal oxide protector was effective to limit over voltage, but a rise in DC bus voltage was noticed. The connection of power electronic load solely limited the overvoltages in LV AC points near the load due to capacitance in the load to buffer the rectifier voltage. The capacitors are very effective at damping fast transient overvoltages. Most DC rectifiers are based on active bridge technologies, it is a small effort to also include protective functionalities such as overvoltage and over-current protections. Mechanical breakers are effective devices, but they have operational delay for noticing the fault and acting. Therefore some passive protective methods could be used such as clamp type protection or addition-al capacitors. Battery also dampened the overvoltage so much that clamp type protection did not even activate. There is uncertainty of battery behaviour in very fast transient phenomena as most of measurements and models do not focus on this, but it is very clear that dampening effect is considerable.

WP3 studied what kind of participation of flexible operation could have to stability of Nordic power system and possible market segments for NPPs regarding the power system stability. Balancing possibilities in FCR-N (frequency containment reserve for normal operation) market was estimated with assumption of 4.6% additional cost of flexible operations. Using market data from 2016, there were 1144 hours when flexible operation could have been profitable. In this case 4.6% increase was calculated respect to Nordpool SPOT price, but in reality there is plant specific operation cost. Also a rough estimation of system where all power plants would take part in automatic frequency regulation was done. With that, the capacity factor decrease per plant was estimated to be only 0.5% with two months of measured frequency data with 1 s interval.

Fingrid sees that 2020 onwards rotating generation will be more limited in the power system and there also will be less controllable power plants. This means that price variations might be large. The energy producers currently have better resources than NPP’s for balancing purposes. There has not been instances that Fingrid had to demand nuclear power plant to reduce power or demand disconnection. Market based solutions have been enough for now. For voltage control, however, there has been more requests to change reactive power injection / voltage setpoint. When grid frequency is outside normal operation region, the grid code demands power plant to be controlled lower or higher output linearly respect to deviations in frequency. Instead of bidirectional balancing, nuclear power plants could serve better in down regulation reserve in cases for system over-frequency and normally leave bids to down regulation balancing market. This practice would guarantee down regulation capacity even if NPPs would never win the bids for activation. It should be noted that FCR-D for disaster situations is only defined for situations when there is lack of power in the system (and not for over-frequency). For system stability respect, there are no large risks in NPP participating to balancing. The most obvious risk to system stability is that, if large nuclear plant is taking major role in system balancing and plant disconnects from grid when there is low inertia in the grid (summer time). For risk analysis perspective, role of single plant in balancing should be limited. It is likely that pressures on all generation to participate more actively on system balancing will increase, and it is very likely that new NPPs will be required to take part at some point of their long operation life cycle.

The project outlined also a risk analytic approach to assess options for flexible operation. The optimization of operational strategies of NPPs is both a multi-criteria task and an issue for multiple stakeholders. Just looking at from a single unit point of view is not sufficient, but it should be analysed from the portfolio of generating units point of view. It is suggested that the assessment can be divided into three major categories: 1) grid system risks, 2) economical risks (of the plant owner), and 3) NPP reactor safety risks. These categories can be broken down into subcategories that can be assessed separately, see Figure 2.14. This approach leads to a multi-attribute decision analysis framework, which also allows to take uncertainties into consideration.

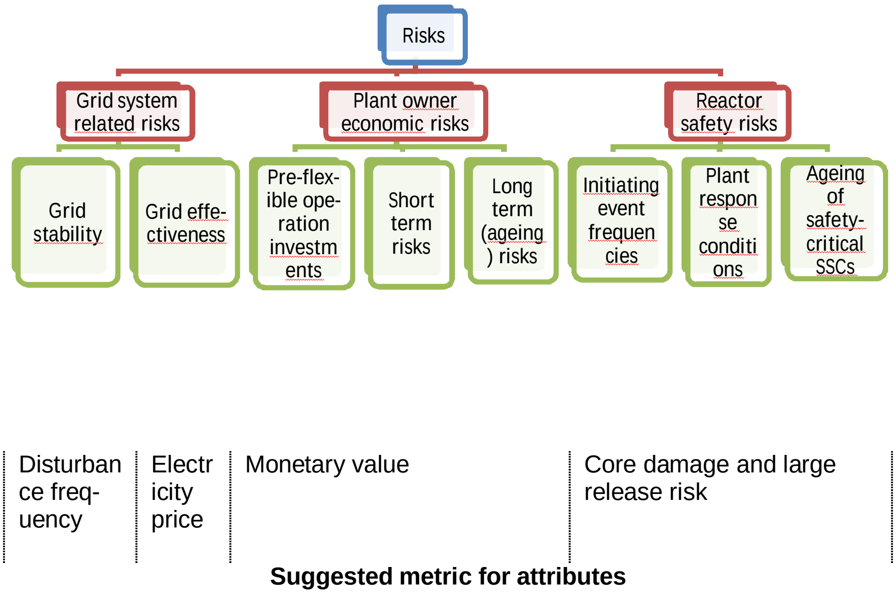


Figure 2.14. Multi-attribute decision making structure.

Deliverables in 2018

- A conference paper in EPE 2018 (Electric Power Engineering) conference describing analysis of open phase condition influence on an induction motor.
- A conference paper was submitted to IEEE PES PowerTech Conference based on open phase condition scenarios for NPP
- Research report was published from unbalances caused by different OPC cases in NPP electric systems and preparedness of operating plants against OPC.
- Research report was published from methods for detection of the OPC condition and recommendations to improve safety of NPPs in the case of OPC.

- A scientific article describing the performance of large-scale grounding systems of NPP against lightning strikes in the IEEE Transactions on Electromagnetic Compatibility.
- A conference paper was submitted to EPE 2019 conference based on lightning overvoltages in electrical power system of a NPP.
- A conference paper was submitted to EPE 2019 conference describing the protection against ground potential rise and lightning overvoltages in control systems of NPP.
- Research report was published from power plant lightning overvoltage protection of low voltage power electronics.
- Master's thesis on lightning induced over-voltages in nuclear power plants.
- Research report was published from risks of adaptive control of NPP electrical systems and stability of the grid.
- Research report on the preliminary risk analysis for adaptive operation of NPP

2.1.8 ORSAPP - Practical applications and further development of Overall Safety Concept

Overall safety is topical in the Finnish nuclear community's parlance and this overall safety and robustness are approached from organizational factors' and institutional strength-in-depth (ISiD) perspectives. The study consists of two tasks. The first task deals with the robustness of Finnish nuclear community in the context of decommissioning that is relevant as regards overall safety. The analysis provides insights into strengths and possible weaknesses in the preparatory phase of the decommissioning and indicates how the robustness could be developed further in organizational aspects. The second task examines the robustness of licensee in the context of incident investigations and learning from them. The study examines incident reports and their underlying rationality, based on which the licensee organization approaches the incidents and learns from them. The latter analysis illuminates organizational rationality in the context of incident investigation and contributes to the organization's possibility to learn more.

Specific goals in 2018

The goal of the study was to examine the robustness of Finnish nuclear community in the context of cases related to decommissioning and robustness of license holder in the context of incident investigations. The concept of institutional strength-in-depth, introduced by the IAEA 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 data of the decommissioning case consisted of decommissioning reports and interviews with experts from the Radiation and Nuclear Safety Authority (STUK), representatives from the Ministry of Economic Affairs and Employment (MEAE), and experts from a power company. Despite the small number of interviewees, the study provided a sectional view over the decommissioning and the relationships between the organizations.

Thematic interviews were exploited as method of data gathering in the decommissioning case. Themes cover the main challenges related to decommissioning, cooperation between the organizations, expectations related to other organizations, funding mechanisms, attitudes concerning participation in commercial decommissioning. Interviews were first transcribed into files, and then they were classified based on the themes. After that, meanings related to the decommissioning were analysed, and underlying rationality of each theme was examined. Interpretations of interviews draw on studies on governance and the concept of ISiD. Qualitative content analysis was exploited as a method of analysis.

The data of incident investigation case consist of 19 incident reports and interviews with three experts from the power company. The goal of the licensee organization is to collect information also about less important incidents and to learn from them. Therefore, incident reports provided by the company concerned cases, which were minor in terms of safety.

The incident reports were analyzed based on the content analysis. First, all reports were written in the form of table and classified based on the following criteria: operation condition, safety class, safety effect, immediate cause of the incident, contributing factors, description of the event, lessons learned. Second, the main reasons for incidents were collected from the reports. Third, the underlying rationality related to incident investigations and learnings from incidents, particularly related to handling of communication and organizational factors, was examined.

Summary and conclusions concerning decommissioning

Meanings related to decommissioning are relevant because they tell about motivation of actors as regards decommissioning. Experts of the Ministry of Economic Affairs and Employment (MEAE) emphasized reputation and credibility of Finland as a country of high technology and Posiva as the world’s first disposal for spent nuclear fuel. These factors create pressure to succeed in decommissioning. Furthermore, representatives of the MEAE associated successful handling of decommissioning with public acceptability of nuclear power. The MEAE aims to create positive image about the decommissioning by referring to abovementioned aspects and Finnish sense of responsibility and possibility to participate in decommissioning business.

Experts of Radiation and Nuclear Safety Authority, STUK, approached decommissioning from the pragmatic viewpoint. They stressed continuous learning process, need to update the requirements and need to create own solutions and decision-making as regards decommissioning due to the lack of common guidelines for decommissioning in the international level. Difficulties to create common international guidelines for decommissioning is because each country has its own practices and ways to implement decommissioning.
Representatives from one power company were interviewed that means that the meanings are indicative, not representative to all power companies. The power company expressed its active and responsible role in planning of decommissioning. It stressed its good preparedness for decommissioning, and it trusted in its own organization and plans.

From the point of view of overall safety, there is a danger that knowledge related to the critical points and factors in decommissioning is shared by small group of people but that knowledge is not spread throughout the organization. There is a need for holistic understanding of decommissioning and better integration of different expertise into planning of decommissioning. For instance, economy, safety, technical and organizational aspects shall be dealt with as intertwined factors. This is important from the overall safety perspective. In the final report, recommendations were provided as regards organizational aspects in decommissioning. Similarly, from the overall safety perspective, the comparison of cost estimations with the foreign decommissioning projects is seen relevant in order to evaluate the sufficiency of VYR funding.

Finally, due to that fact that there are different ways to implement decommissioning and the international guidelines for decommissioning are lacking, the role Finnish nuclear community, and all stakeholders are emphasized. Therefore, stakeholders' mutual cooperation is important.

Summary and conclusions concerning incident investigations

Robustness of organization i.e. its institutional strength-in-depth can be assessed in relation to organization's approaches to incident investigations and ability to learn from incidents. Incident reports show that the instrumental rationality is organization's principal way to handle and govern incidents and learn from them. Instrumental rationality manifests itself in the form of procedures, such as accimap, which provides a general framework for investigations. Accimap focuses on causes and effects, but is not ideal for illustrating organizational aspects. Similarly, tools like phonetic alphabets are good practical tools but they do not help if there are deeper problems related to social interaction in organization. Therefore, if organization aims at deeper and broader learning, Instrumental rationality is not an adequate approach. By resorting to instrumental rationality, incident reporting and learning from them remains thin. Inadequate learning from incidents is noticed in the studies on safety critical organizations in general.

License holder has actively developed its incident investigation practices and learning from incidents. Willingness to learn from minor incidents shows license holder's ambition and motivation. Instrumental rationality way of governing incidents is prevailing and it has provided good pragmatic tools. However, instrumental rationality cannot identify patterns that would require understanding of broader social context. Therefore, new perspectives and approaches, such as reflexive responsive rationality, would be needed alongside with the instrumental rationality, to improve incident investigation. In the final report, recommendations and checklist are provided as regards organizational aspects and reflexive responsive process in incident investigation.

Deliverables in 2018

- A research report describing Institutional strength-in-depth in the context of decommissioning and incident investigations.

2.2 Reactor safety

In 2018 the research area "Reactor safety" consisted of eleven projects:

1. Comprehensive analysis of severe accidents (CASA)
2. Chemistry and transport of fission products (CATFIS)
3. Comprehensive and systematic validation of independent safety analysis tools (COVA)
4. Couplings and instabilities in reactor systems (INSTAB)
5. Integral and separate effects tests on thermal-hydraulic problems in reactors (INTEGRA)
6. Nuclear criticality and safety analyses preparedness at VTT (KATVE)
7. Development of a Monte Carlo based calculation sequence for reactor core safety analyses (MONSOON)
8. Development and validation of CFD methods for nuclear reactor safety assessment (NURESA)
9. Physics and chemistry of nuclear fuel (PANCHO)
10. Safety analyses for dynamical events (SADE)
11. Uncertainty and sensitivity analyses for reactor safety (USVA).

2.2.1 CASA - Comprehensive analysis of severe accidents

This project brings together a large spectrum of phenomena related to the thermal hydraulics of severe Nuclear Power Plant (NPP) accidents. The objective is to decrease uncertainties in defining the possible radioactive release to the environment. Analyses also include evaluating the environmental consequences of a hypothetical severe accident. These topics are of high importance in the severe accident management of the Finnish NPPs.

Specific goals in 2018

Work on estimating the coolability of a multi-dimensionally flooded conical debris bed less conservatively establishing a temperature-based dryout criterion was continued. The focus was on the effect of heat transfer and friction models. Fluent and MEWA results were compared to the KTH's DECOSIM analyses. The Fluent simulations with the same friction model as in DECOSIM were further validated with truncated-cone debris beds as the flow field close to the tip with conical bed may hamper qualitative comparison of Fluent results with MEWA and DECOSIM. The differences between Fluent and DECOSIM results showed the same characteristics as obtained previously in the conical bed cases. Implementing the same friction model also into MEWA improved the correspondence of the results notably. In order to obtain the same saturation temperature reported to be in the DECOSIM simulations, the pressure on the top surface was reduced from 3 bar (documented to be used in the DECOSIM simulations) to 1.3 bar. MEWA results for the reduced pressure agree reasonably with the DECOSIM results in all analysed cases.

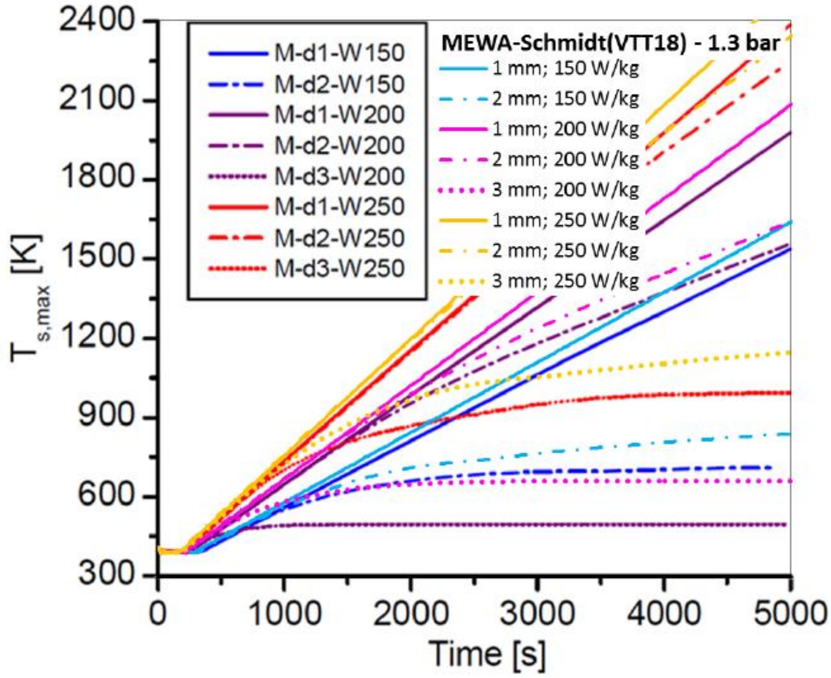


Figure 2.15. Comparison of the time evolution of the maximum solid particle temperature in the MEWA simulations of this study and in the DECOSIM simulations. In the MEWA simulations, the top surface pressure is 1.3 bar.

MELCOR's new core catcher model was tested to enable more reliable reactor application analyses. The WCB-1 experiment was attempted to calculate with the new model. The objective was to test how the new model works and compare the calculation results with the measurements. Unfortunately, the testing of the new model was unsuccessful as the current code revision crashes when attempting to run the calculation. It was concluded that the model is not usable with MELCOR 2.2 revision 11932.

Pool scrubbing experiments performed in the SAFIR2018 CATFIS project provide excellent validation data for the integral codes. The medium-scale pool scrubbing experiments with non-soluble aerosols analysed already in 2017 were re-analysed with corrected mean mass diameter values. Heating the pool was not taken into account in the 2017 analyses. Now, the experiments were analysed with setting the inlet flow to the corresponding pool temperature. In addition, with ASTEC was analysed a case with a nearly stable pool temperature by adding a heat source to the pool. MELCOR results for the Decontamination Factors (DFs) were notably improved due to corrected mean mass diameter values.

ASTEC analyses were also performed for medium-scale experiments with soluble CsI with four different pool depths (30cm, 50cm, 70cm and 90cm) and three differ-ent inlet flow rates for N2 (26.9 l/min, 14.1 l/min and 7.7 l/min). The DF values pro-duced by ASTEC were in a good agreement with the experimental results (Figure 2.16.) Both in the experiment and in the analyses, DF increases with decreasing N2 inlet flow rate and with increasing pool depth. Aerosol balances were also analysed, and it was noticed that the amount of aerosols accumulated in the facility gas phase was notable with lower inlet flow rates.

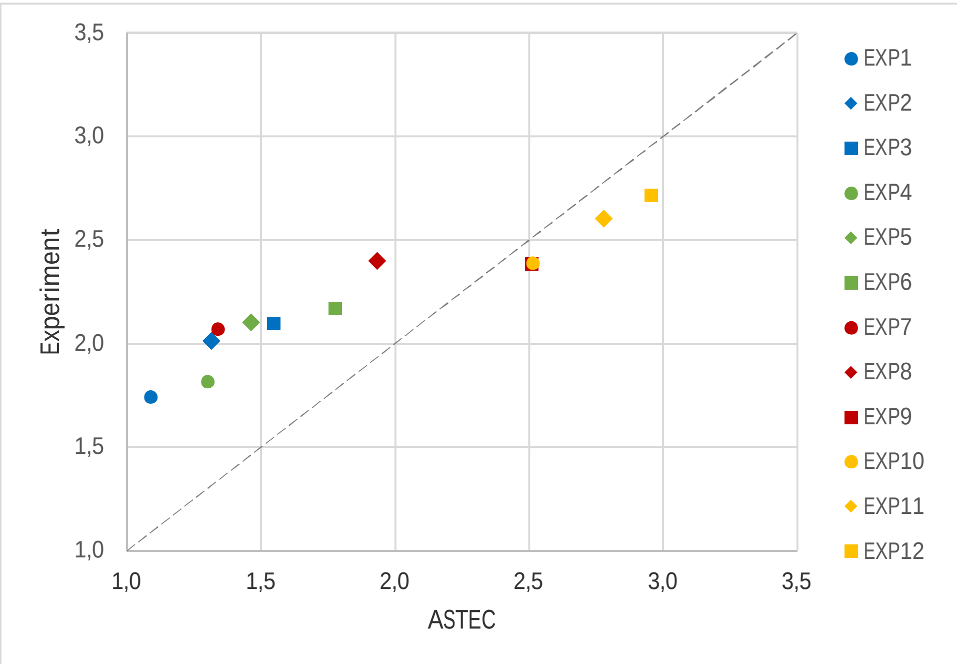


Figure 2.16. Comparison of ASTEC results to the experimental values in the case of non-soluble CsI. Same colour indicates the same pool depth that increases when proceeding to the next phase. Inlet flowrates decrease when proceeding to the next phase for a pool depth.

To complement the understanding on code capabilities to assess the potential source term in long-term severe accident conditions, CsI re-entrainment from water at elevated temperatures was also studied with ASTEC. WH-24 experiment performed in the frame of OECD/NEA THAI-3 project had 12 phases divided into three categories: a) sub-cooled pool, b) boiling pool and c) depressurization. The release seems to increase with the increasing pool temperature. However, with a boiling pool, there are no big differences in the release factors, despite the inlet steam and air flows were varied. With the sub-cooled and boiling pool, ASTEC overestimated the release notably. In the case of depressurization, ASTEC results are not in correspondence with the experimental values.

Modelling practices that can be used in safety analyses of reactors equipped with a Passive Containment Cooling System were developed modelling LUT PCC-10 experiment including aerosol injection to the experimental facility with MELCOR. The experiment was conducted at four different stages, with cooling and ventilation in between and therefore, the analyses was split into four separate calculations. The results of the first stage overestimated the pressure and the cooling ability of the system. As the facility heated up in the following phases, the MELCOR results improved and showed even lower pressures than in the experiments. The MELCOR results also indicated that the PCCS is able to capture aerosols, but further experiments as well as simulations are needed to verify how effective it is.

Understanding of the key phenomena related to hydrogen combustion and flame propagation models in realistic conditions was improved computing the THAI-3 blind benchmark predictions with a CFD-based combustion model developed at VTT for lean hydrogen mixtures. The combustion model was implemented as user defined functions in the CFD-code Fluent. The Zimont model was extended for lean hydrogen mixtures and the flame speed is now assumed to be determined by the dominating phenomenon of laminar burning, flame instabilities, buoyancy and flow turbulence. Furthermore, the influence of the flame thickness on the burning velocity is taken into account for flames with instabilities. The comparison of the blind predictions of the HD-44 results to the preliminary experimental data shows that the computational results are in a reasonable agreement with the experimental data. The maximum pressure is close to the experimental one and the pattern of the flame front propagation is similar enough.

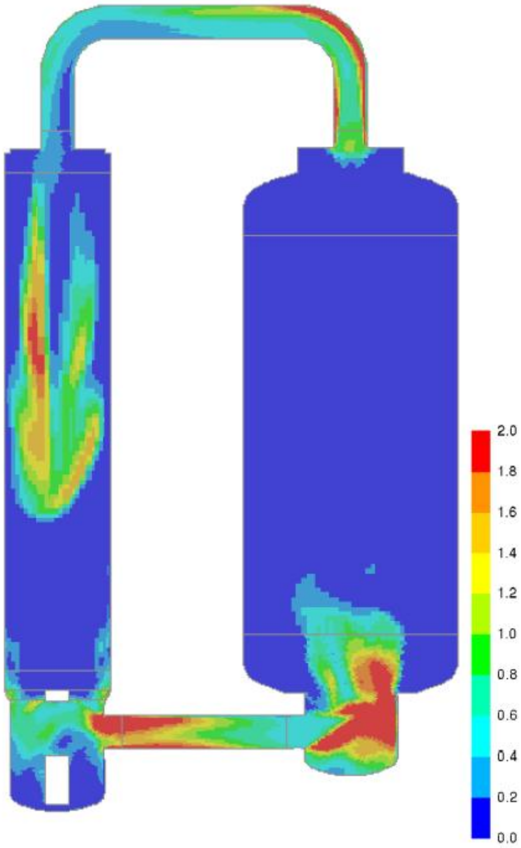


Figure 2.17 Contours of the turbulent kinetic energy (m2/s2) on a vertical cross-section in the HD-44 simulation at the time of the ignition.

However, the flame speed was underestimated by about a factor of 2. The difference is not that unexpected regarding the models determining the flame speed in these conditions.

After previous years' developments, the validity of VALMA-calculated doses at long distances was evaluated comparing them not only to ARANO but also to MACCS, that is a widely validated code developed by Sandia National Laboratories. MACCS was developed as a general-purpose tool applicable to diverse reactor and non-reactor facilities. The principal phenomena included in these three codes are atmospheric transport and deposition under prevailing meteorology, short- and long-term mitigation actions and exposure pathways, deterministic and stochastic health effects and economic costs. Implementation of MACCS demonstrates that this internationally well-known code is available to be used at VTT and capable to calculate offsite radiation doses.

The calculated cases dealt with offsite doses without countermeasures in a single weather condition as well as with the probabilistic approach employing annual weather data.

Calculation results indicate significant differences in some single atmospheric dispersion cases, but differences are reduced if annual weather is used and fractile (percentile) values are presented. Then ARANO typically predicts smaller dose values than MACCS. Also comparable dose estimates of VALMA predict smaller dose values than MACCS. The results may indicate that MACCS, a code with long history, was coded with some conservatism in mind.

Deliverables in 2018

- VTT's results of the Fukushima unit 1, 2 and 3 MELCOR calculations and descriptions of the calculation models were submitted to the BSAF-2 operating agent. The draft final report was reviewed and comments were sent.
- A presentation was given in the CSARP/MCAP meeting on severe accident analyses for the license application review of OL3. A travel report summarizing the most interesting presentations was written.
- MEWA and Fluent simulations were performed for truncated-cone debris beds and the results were compared with KTH's DECOSIM analyses. MEWA was implemented with the same friction model as DECOSIM and Fluent earlier.
- MELCOR's new core catcher model was tested. As the current code revision crashes, the features of the new core catcher model and the attempts to use the model were described.
- The PCC-10 experiment with aerosol injection was calculated with MELCOR. The analysis was split into four different parts, due to the ventilation of the test facility.
- In total 12 pool scrubbing experiments with soluble CsI were analysed with ASTEC. Also the experiments with non-soluble aerosols analysed already in 2017 were re-calculated with ASTEC and MELCOR with corrected mass median diameter values.
- VTT's combustion model for hydrogen combustion and flame propagation was revised. Computational meshes were created for the THAI-3 benchmark and four test cases. The updated model was tested in five cases. Blind simulation for the benchmark was performed.
- All 12 phases of the WH-24 experiment to study CsI re-entrainment from water at elevated temperature were analysed and the results were presented in the 5th PRG meeting.

The results calculated by the offsite dispersion and dose assessment codes ARANO, VALMA and MACCS were compared.

2.2.2 CATFIS - Chemistry and transport of fission products

The objective of the project (2015-2018) is to study the behaviour of fission products in severe accident conditions. In particular, the aim is to increase understanding of revaporisation and transport of iodine in primary circuit and containment of a nuclear power plant. The primary circuit study has been conducted in close co-operation with IRSN Cadarache research centre for the determination of iodine chemistry. The objective of the primary circuit study at VTT is to determine iodine compounds released due to the reactions on the surface of primary circuit piping. At the same time IRSN is focused on the gas phase chemistry of iodine in similar experimental conditions. The measurements with the EXSI-PC facility of VTT provide information on high temperature chemistry and facilitate validation of for example iodine chemistry codes. The second aim is to find out the effect of primary circuit conditions on the transport and speciation of ruthenium. These experiments are conducted with VTT´s Ru transport facility in collaboration with Chalmers University of Technology as part of NKS-R activity. As a third aim, radiolytical reactions by various radiation sources in containment conditions is studied using EXSI-CONT and BESSEL facilities. The objective is to verify the possible oxidation of iodine into particles and also the formation of nitric acid. The fourth aim is to study the pool scrubbing behaviour of fission products in the water pools of containment building. The objective is to widen the knowledge towards conditions which are not well-known and to enhance source term calculations. In addition, the gathered data in all experiments is used to derive models for the studied reactions, which can eventually be implemented in severe accident analysis codes.

International collaboration is conducted by participation in the work of OECD/NEA STEM-2, OECD/NEA BIP-3 (both started in 2016) and NUGENIA programmes. The data of experiments performed as part of SAFIR2018 will be shared within these forums, as well as information related to the progress of programmes will be distributed to SAFIR2018 members.

Specific goals in 2018

The main goal in 2018 was to study the effect of reactions of iodine containing deposits on primary circuit surfaces on the release and transport of iodine. Fission product deposits on primary circuit surfaces can act as a source of gaseous iodine even in the late phase of a severe accident. However, that is not sufficiently considered in the severe accident analysis codes currently. The primary circuit experiments were conducted using the updated EXSI-PC facility.

The second goal was to study the formation of nitric acid (HNO_3) by radiation. The aim was to verify the capability of beta radiation to produce nitric acid in humid air simulating containment conditions in a severe accident. The objective was to compare the results with the previous gamma radiation results. The third goal was to study the effect of HNO_3 generated by beta radiation on the pH of containment pools. This was performed utilizing the data of experiments (in the above task) in ChemPool calculations. The effect of beta radiation e.g. on the formation of nitric acid is currently poorly known, although beta decay corresponds for a significant fraction of the accumulated radiation dose in the containment atmosphere. As a fourth goal the retention of aerosols and gaseous organic iodine in the containment pool was studied. The experiments were performed at pool temperatures up to boiling point, since the existing data is mainly limited to studies at 20 °C. The performed aerosol experiments were simulated with ASTEC and MELCOR codes by CASA project and the experimentally and analytically obtained decontamination factors were compared. This task is also connected to the NUGENIA TA2.4 area IPRESCA project dedicated to pool scrubbing research internationally.

Another goal was to continue the follow-up of the OECD/NEA STEM-2 and OECD/NEA BIP-3 programmes. The durations of programmes are four years and three years, respectively. The programmes will e.g. verify the findings of the CATFIS project in ruthenium chemistry, and also produce complementary and new data on ruthenium transport in the RCS and on iodine behaviour in the gas phase of containment building and on the painted containment walls. A significant part of the programmes is to compare the performance of various severe accident analysis codes and the user effect.

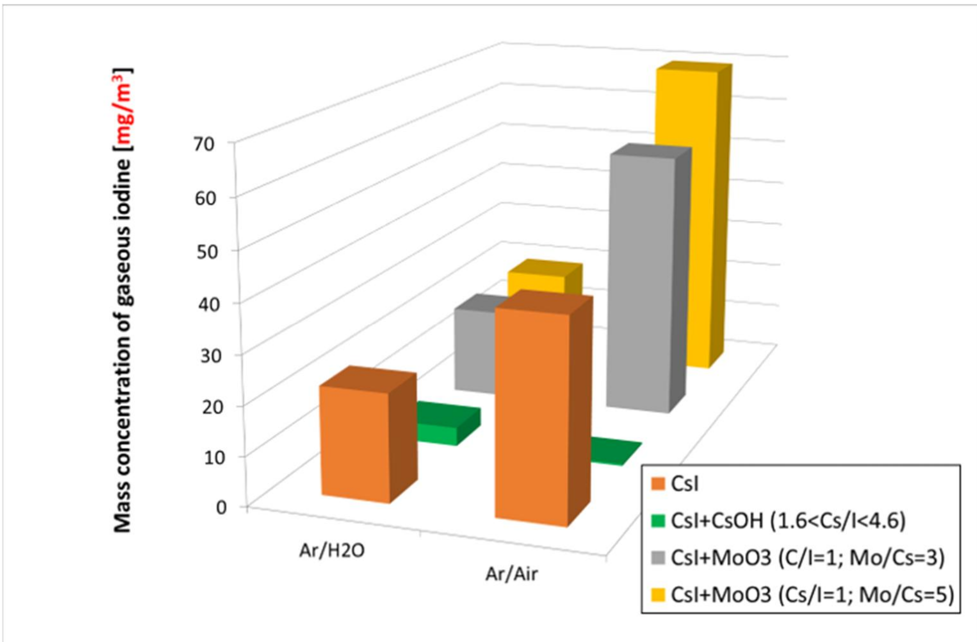


Figure 2.18. Gaseous iodine transport [mg/m^3] as a function of the precursor and carrier gas compositions. The experiments were performed at 700 °C. The addition of molybdenum to caesium iodide resulted in a higher gaseous iodine fraction in an atmosphere composed of argon and air. For all the studied initial Mo/Cs ratios and initial precursor masses, the gaseous iodine release was higher when the oxygen partial pressure was higher (i.e. for Ar/Air atmosphere). The formation of caesium molybdate was identified in the evaporation crucible after the experiments, confirming that the reaction between caesium and molybdenum is the reason for the observed formation of gaseous iodine. Gaseous molecular iodine (I_2) was clearly identified as the main gaseous iodine species formed during the reaction (> 68 %). On the other hand, the addition of caesium hydroxide to caesium iodide seemed to result in a decrease in the gaseous iodine fraction in both Ar/H_2O and Ar/Air atmospheres.

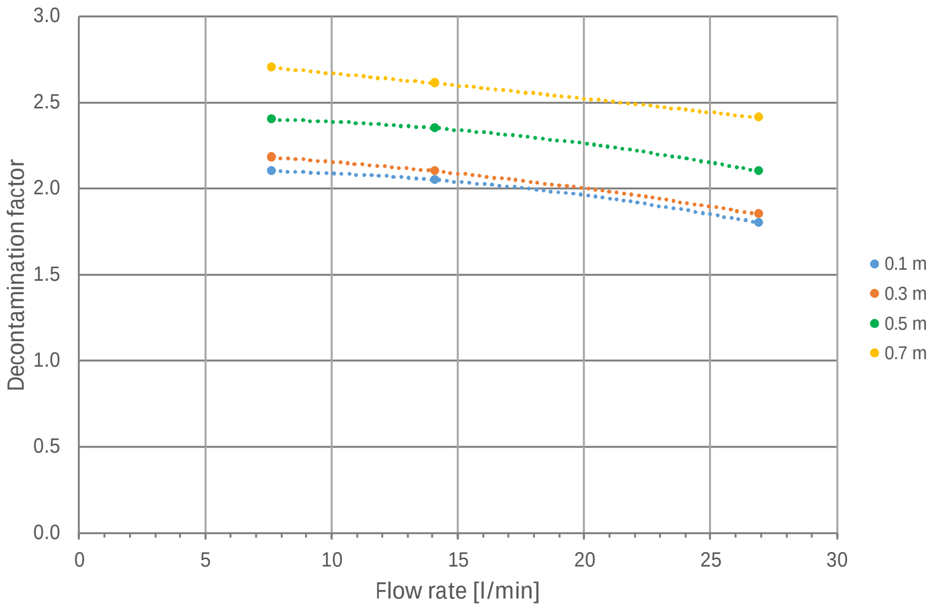


Figure 2.19. The retention of CsI aerosol in a water pool with a height of 0.1, 0.3, 0.5 and 0.7 m from the nozzle to the pool surface was examined at 20 °C with the medium scale pool facility. The geometric mean of the fed CsI aerosol mass size distribution was 0.2 µm. The nitrogen gas flow rate to the pool ranged from ca. 8 to 27 l/min. The preliminary decontamination factor (DF) results of these experiments are presented as a function of flow rate. DF seemed to decrease due to increasing flow rate through the pool. On the other hand, the deeper pool was resulting in a higher DF.

Deliverables in 2018

- In the primary circuit studies the source of iodine was CsI powder which was evaporated at 700 °C on ceramic surface under Ar/H_2O and Ar/Air atmospheres. The surface of the reaction furnace tube, made of stainless steel, was pre-oxidized before the experiments. Two mixtures of CsI with MoO_3 (molar ratios Mo/Cs = 3 and Mo/Cs = 5) have been tested for each atmosphere composition. To summarize the main outcome of experiments, a notable, increased release of gaseous iodine from CsI powder was observed at 700 °C when molybdenum was mixed with caesium iodide as a precursor, see Figure 2.18. A conference paper for the International Congress on Advances in Nuclear Power Plants (ICAPP 2018) was written and presented.
- A scientific publication concerning the studies of cadmium and silver effect on the iodine chemistry in primary circuit was submitted to Nuclear Materials & Energy.
- The formation of nitric acid by beta irradiation in humid air simulating containment conditions in a severe accident was experimentally verified. The experimental results were used to determine a G-value (radiolytic-yield value, molecules formed per 100 eV of energy absorbed) for the formation of HNO_3 by beta radiation. As a result, the G(HNO_3)-value was determined for the first time considering beta radiation in the containment gas phase. The result was compared with gamma radiation in similar conditions, it seemed that beta radiation resulted in a slightly higher G-value, although considering the uncertainties of experiments both values are practically the same. The obtained new G-value was utilized in the ChemPool code analysis on the containment pool pH evolution in a severe accident. The investigated accident scenario was a total blackout and a big break LOCA in the cold leg of a PWR. The new G(HNO_3)-value resulted in a significant decrease of pH to a level of pH 2 to 3 in the pools of containment. As pH drops the formation of molecular iodine (I_2) begins in the pools. In acidic conditions, volatile form of radioactive iodine could be formed and released into atmosphere. This result was significant and the potential release of iodine from the pools was indicated.
- Two experimental setups simulating small and medium scale pools have been built previously to study NPP containment pool efficiency in the retention of fission products as aerosol particles and gaseous compounds. The experimentally determined decontamination factor (DF) increased with the increasing pool depth (up to 70 cm pool depth used in the experiments) and decreased with the increasing air or N_2 flow rate through the pool (see an example in Figure 2.19). In general, integral codes resulted smaller DFs than recorded in the experiment (simulations performed in CASA project). This is conservative what comes to the potential source term. MELCOR simulations were in a good agreement with the experiments about the DFs for non-soluble aerosols. ASTEC V2.1.1.4 resulted in quite similar DF results in comparison to experimental results for soluble CsI aerosol. This study will be continued in the coming years to improve understanding on the pool scrubbing phenomenon. The experimental and analytical methods will be further enhanced to allow the code validation covering low/high temperatures and flow rates etc.
- The second meeting of NUGENIA TA2.4 area IPRESCA project dedicated to pool scrubbing research was participated. CATFIS is coordinating a task on gaseous iodine retention in the pool. The project includes over 30 international participating organizations around the world.
- The effect of air radiolysis products (N_2O, NO_2, HNO_3) on ruthenium transport was studied in SAFIR2014/SAFIR2018 programs as part of TRAFI and CATFIS projects. Currently, the NO_2 effect on Ru transport is also a research topic in the ongoing international OECD/NEA STEM-2 programme. The STEM / START experimental facility is very similar to the experimental facility of CATFIS. CATFIS project was asked to perform complementary experiments for the STEM-2 programme. As part of experiments, the increased transport

of gaseous RuO_4 was studied in more detail. The evolution of RuO_4 transport rate during the experiment was obtained with the UV-Vis analysis method.

2.2.3 COVA - Comprehensive and systematic validation of independent safety analysis tools

The COVA project aimed at developing and promoting a rigorous and systematic approach to the procedures utilized in validation of independent nuclear safety analysis tools. The process enhanced 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. Main part of the work was carried out with the system-scale safety analysis tool Apros that has been developed in Finland in cooperation between VTT and Fortum and that is currently used in safety analysis work both at the regulatory side and by Finnish utilities Fortum and TVO. The U.S. NRC's TRACE code that is currently used by VTT for the Finnish regulatory body STUK provides suitable benchmark in the validation process as an independent, widely used and well validated safety analysis tool. Participation in international research projects related to nuclear safety research in the field of thermal hydraulics formed an essential part of the project: experimental data produced in these activities was directly utilized in the validation work carried out within COVA, and on the other hand, these validation activities supported conduction of the experiments, in addition to promoting international cooperation and networking in the field of nuclear safety research.

COVA was divided into four work packages: Validation matrices, Analyses of new experiments, Management and international cooperation and Participation fees. The actual research work dealing with analysis tool validation was carried out in the first two work packages, with the first one concentrating on the fundamental aspects of the validation work with Apros, and the second in application of Apros and TRACE to validation using primarily integral-scale experiments. Third work package contained all the administrative work in the project and all costs arising from participating in the international projects and reporting of their results to the Finnish research community, with the exception of the participation fees. The fourth work package included the participation fees of international research projects and nothing else.

Specific goals in 2018

Assessment of Apros' validation base was performed in the first year of COVA and in 2018 analyses were performed in order to fill the validation gaps identified in the TH and containment model's assessment reports.

Three film boiling experiments, performed in the Pennsylvania State University/U.S. Nuclear Regulatory Commission Rod Bundle Heat Transfer (RBHT) Test Facility, were simulated with Apros. The study was carried out as a summer job.

TOSQAN spray test 101 was conducted with the 6.09- base trunk version of Apros. The goal of the work was to test the multi-size droplet modelling capability of the Apros containment code against the experimental data.

A number of Becker critical heat flux experiments with 3 and 7-rod clusters were simulated with Apros.

The PKL4 i2.2 run3 benchmark was simulated in Apros in order to assess system code capabilities regarding a 17 % IB-LOCA scenario.

Apros model of the PASI test facility, located in Lappeenranta University of Technology, was created (Figure 2.20). The natural circulation loop of the facility was modelled using the 6-equation model of Apros and the containment pressure vessel was modelled using the containment package of Apros. Several pressure loss experiments were calculated using the Apros 6.07.34.07 version and the results were compared to the experiment measurements. The findings of the comparison were used to tune the pressure losses of the Apros model so that the calculated pressure losses were in line with the measured values.

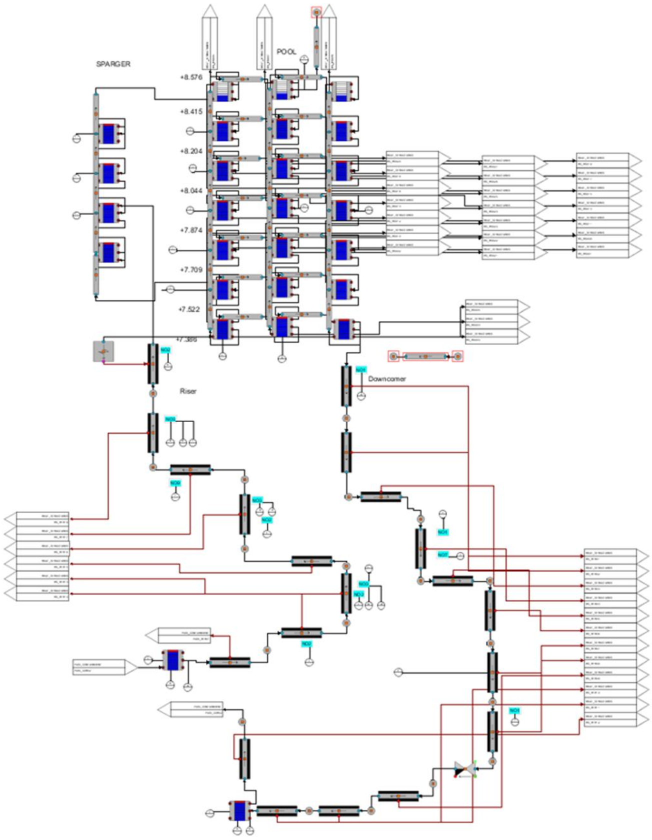


Figure 2.20. Apros model of the PASI test facility.

Five international cooperation programmes were followed in COVA project. These were OECD/NEA PKL-4, HYMERES and WGAMA, U.S. NRC CAMP and FONESYS network. Participation fees were paid for OECD/NEA HYMERES Phase 2 and USNRC CAMP.

Deliverables in 2018

- A report on Becker critical heat flux experiments
- A report on RBHT reflooding heat transfer experiments
- A report on TOSQAN spray test 101
- A report on PKL4 benchmark
- A report on PASI characterizing experiments
- A report on BNL critical flow experiments

2.2.4 INSTAB - Couplings and instabilities in reactor systems

The INSTAB project aims to increase understanding of the phenomena related to BWR pressure suppression function to enhance capabilities to analyse Nordic BWR containments under transient and accident conditions. Particularly, additional information is needed on the effect of Safety Relief Valve (SRV) spargers, residual heat removal (RHR) system nozzles, strainers and blowdown pipes on mixing and stratification of the pool as well as feedbacks between wetwell water pool and spray i.e. formation and mixing of thermally stratified water layers in the suppression pool due to spray operation. A combined experimental/analytical/computational program is carried out where Lappeenranta University of Technology (LUT) is responsible for developing an experimental database on pool operation related phenomena in the PPOOLEX and SEF-POOL test facilities with the help of sophisticated, high frequency measurement instrumentation and high-speed video cameras. LUT, VTT and KTH will use the gathered experimental database for the development, improvement and validation of numerical simulation models. The project outcome will allow the end users to analyse the risks related to different scenarios of safety importance in the drywell and wetwell compartments of a Nordic BWR.

Specific goals in 2018

Specific goals in 2018 included a test with the SRV sparger in the PPOOLEX facility after the sparger first had been moved to the center position in the pool. In addition, the sparger head was moved up and the pool water level was increased in order to increase the thickness of the cold stratified layer. SRV sparger studies were also conducted with a small-scale separate effect test facility (SEF-POOL) enabling a direct measurement of force induced by steam injection. The main motivation for all these tests was to support the development and validation work of the Effective Heat Source (EHS) and Effective Momentum Source (EMS) models being done at KTH. Experiment results obtained in the INSTAB project are extensively utilized also by LUT and VTT in parallel SAFIR2018 projects for the development/validation work of direct contact condensation (DCC) models to be used in CFD codes such as NEPTUNE_CFD and OpenFoam. A review of published works concerning the studies of the release and dissolution of non-condensable gases (NCGs) and their possible effect on the reactor coolant circuits was also carried out.

The pressure suppression pool in a BWR serves as a primary heat sink during a loss of coolant accident (LOCA) or in case the reactor is isolated from the main heat sink. The pool surface temperature defines the saturation steam pressure in the containment. Steam condensation creates a source of heat in the pool. In case of small steam flow rates, thermal stratification could develop and significantly impede the pressure suppression capacity of the condensation pool. Experimental studies have shown that once steam flow rate increases significantly, momentum introduced by the steam injection and/or periodic expansion and collapse of large steam bubbles due to DCC can destroy stratified layers and lead to mixing of the pool water. Accurate and computationally efficient prediction of the pool thermal-hydraulics with thermal stratification, mixing, and transition between them, presents a computational challenge.

KTH is developing the simplified EHS and EMS models and implementing them the in GOTHIC and ANSYS Fluent codes. The models aim to capture thermal stratification and mixing phenomena in a large pool of water. They can be implemented also in system codes, such as APROS. The models have already been validated against PPOOLEX experiments where the dynamics of free water surface in the blowdown pipe with different steam mass flow rates and transient times was studied. KTH is now extending the validity of the EHS and EMS models to spargers, RHR nozzles and sprays.
In 2017, the SRV sparger in the PPOOLEX test facility was moved to a centre position in the pool. Originally, it was about 0.42 m away from the pool centre axis. In 2018, the sparger piping was shortened so that the distance of the sparger head from the pool bottom increased from 1200 mm to 2000 mm. Furthermore, the pool water level was increased 3300 mm. The goal

was to have a thicker cold stratified layer to enable a more efficient analysis of the relation between the erosion velocity and the distance between the sparger and thermocline. A sparger test was then done and compared to earlier tests to find out how the change in the sparger position and elevation affects stratification/erosion/mixing behaviour during steam discharge. Particularly, the effect on the elevation and thickness of the thermocline between the cold and warm water volumes and on the temperature profile of the pool were of interest. According to the direct contact condensation mode map for pure steam discharge of Chan and Lee, the dominant flow mode in the 2018 test and in the earlier reference tests was oscillatory bubble (Figure 2.21).

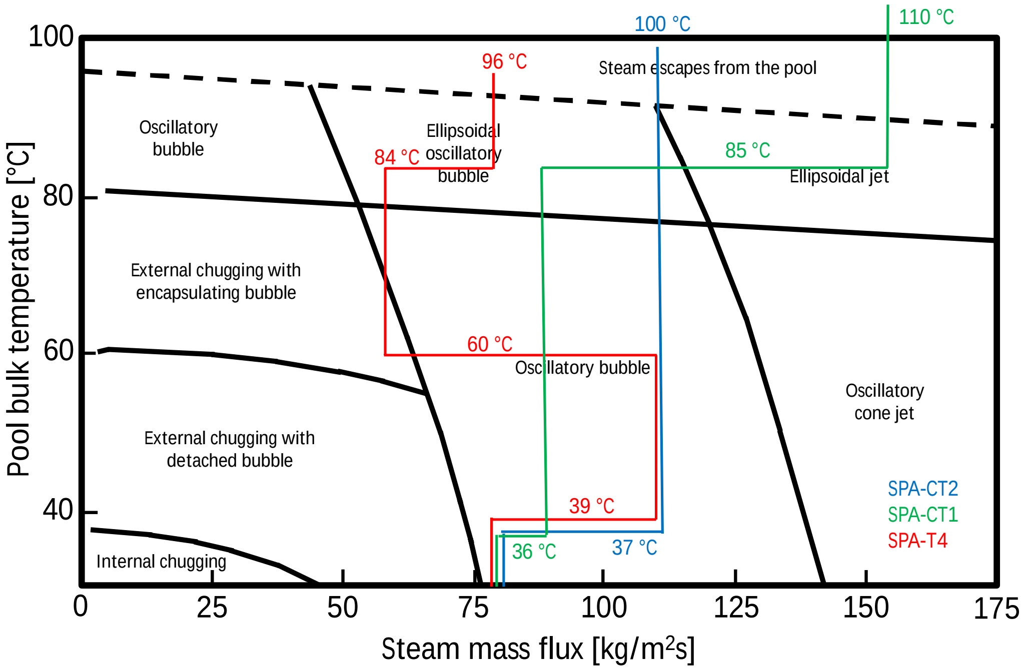


Figure 2.21. Paths of the SPA-CT2, SPA-CT1 and SPA-T4 sparger tests marked on the direct condensation mode map for pure steam discharge of Chan and Lee.

During the erosion phase, the elevation of the thermocline shifted slowly downwards. However, complete mixing was not achieved and eventually the erosion process slowed down and some kind of equilibrium situation developed for a while. Then, a new stratification process started although the steam mass flow rate was kept constant (Figure 2.22).

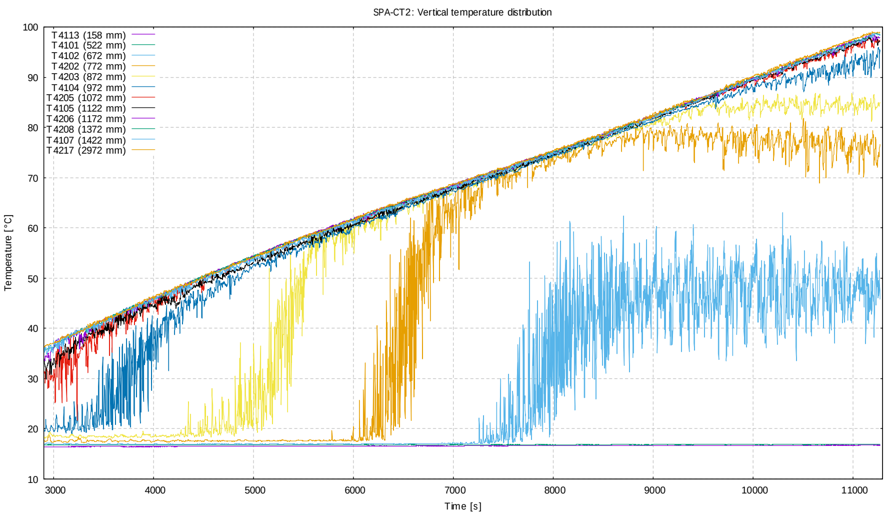


Figure 2.22. Vertical temperature distribution in the wetwell pool during the erosion phase (2940-11200 s) in the SPA-CT2 test with the sparger in the centre position.

The SEF-POOL facility was designed together by KTH and LUT. It was constructed by the nuclear engineering research group at LUT and it is particularly used for the validation of the EMS model proposed by KTH for simulation of steam injection into a pool filled with sub-cooled water. The design of the facility is such that the effective momentum (liquid force carried by the condensate liquid) can be directly measured with a force sensor or it can calculated on the basis of measured steam momentum (steam force at the injection hole).

The goal in the SEF-POOL tests is 2018 was to obtain data of the characteristics of small-scale phenomena affecting the effective heat and momentum sources. Steam mass flux, sub-cooling and injection hole diameter were the variables affecting the condensation regime, which were analysed during the tests. Momentum was measured with force sensors and high-speed video clips of the DCC phenomenon were recorded. The oscillatory bubble, partly the oscillatory cone jet and partly the stable jet regimes were the covered flow modes.

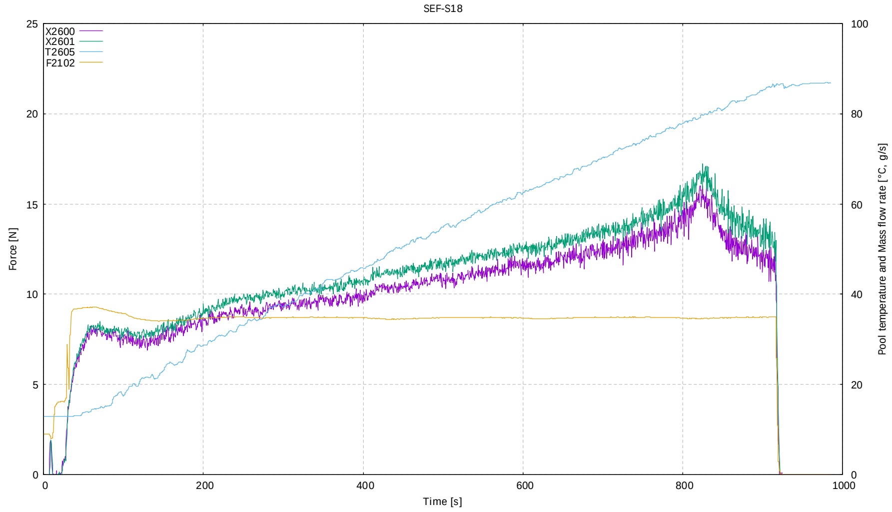


Figure 2.23. Measured forces (X2600, X2601), flow rate (F2102) and pool temperature (T2605) in a SEF-POOL test.

All sub-sonic regimes presented a strong dependency of measured forces with the sub-cooling, whereas this effect was practically negligible in stable jet regimes occurring above 300 kg/m2s. However, force sensors started to indicate a clear decreasing trend after the pool water temperature exceeded 80 °C (Figure 2.23). This behaviour can be attributed to the flow mode change to ellipsoidal jet regime.

Qualitative analysis of video images show that the oscillatory motion of the steam bubbles can be divided into three parts. That is, the bubble begins to grow attached to the injection hole, detaches when the force balance becomes positive in the direction of the steam injection, and collapses as the neck connecting the bubble to the injection hole reduces the steam flow into the bubble (Figure 2.24). At low subcoolings, detached bubbles can move a large enough distance from the injection hole and allow the formation of a new bubble before their collapse.

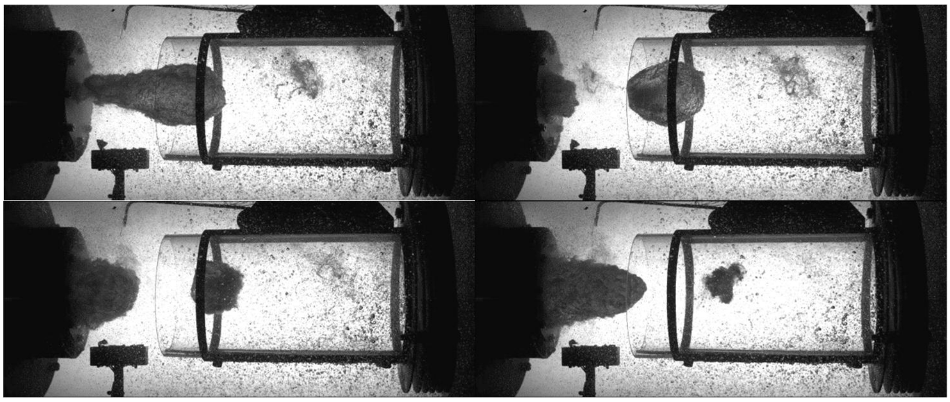


Figure 2.24. Video frames from a SEF-POOL test (steam mass flux 174 kg/m2s, pool water temperature about 85 °C, injection hole diameter 16 mm, Δt=4285,68 μs).

Bubble parameters, which have been estimated through image processing of the high-speed videos by KTH, include for example the collapse and bubble life frequencies, maximum bubble radius and bubble velocities. Figure 2.25 presents examples of the obtained bubble collapsing frequencies. The collapsing frequencies present a quasi-linear dependency with the sub-cooling. In the low steam mass flux tests S10 and S14, the frequency stabilizes when the sub-cooling is above ~60 °C.

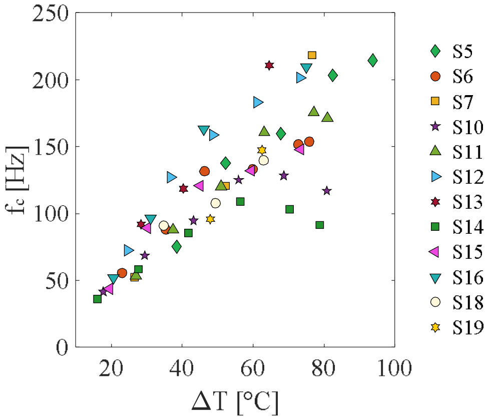


Figure 2.25. Bubble collapsing frequencies as a function of sub-cooling in the SEF-POOL tests.

The presence NCGs and their corresponding effects in light water reactor (LWR) coolant system are of special importance in the analysis of thermal-hydraulic behaviour of the nuclear power plants (NPPs). For example, the existence of NCG can obstruct the coolant path and interfere in the natural circulation and it can decrease heat transfer significantly to the secondary circuit and may thus prevent proper cooling of the core. A literature survey concerning the studies of the release and dissolution of NCGs and their possible effect on the reactor coolant circuits was conducted in 2018.

The published works on the release and dissolution of NCGs are rather sparse despite of the large role played by NCGs in reactor coolant system. Very few works (mostly experimental and analytical) have been done on this issue. There is no any published work found on the release and dissolution of NCGs with computational fluid dynamics (CFD) approach. On the other hand, many works concerning the effects of NCGs in reactor coolant system have been performed by experimentally, theoretically and numerically.

The effects of dynamics on NCG dissolution and release are at present not completely understood. Improving the modelling of NCG dissolution and release is of interest for all safety analyses, containment and reactor. Traditionally, multicomponent modelling has assumed that NCGs travel only with the steam phase. However, in reality, dissolution in water masses and release, also takes place. However, in this area, still the applicability of modern CFD codes and modern measurements techniques (e.g. PIV, high speed video recording) could be very valuable.

Deliverables in 2018

- The SRV sparger in the PPOOLEX test facility was shortened and the pool water level was increased in order to increase the thickness of the cold stratified layer. The effect on stratification/erosion/mixing behaviour during steam discharge via the sparger pipe was studied in a test and compared to reference tests.

- An extensive test series was carried out in the SEF-POOL separate effect test facility. Data of the characteristics of small-scale phenomena affecting the effective heat and momentum sources was provided to be used for the validation of the simplified EMS/EHS models proposed by KTH. The SEF-POOL results also supported the validation effort of the DCC and interfacial area models of CFD codes for steam injection through spargers at VTT and LUT.

- A literature survey of existing experiment data and models for noncondensible dissolution/release dynamics and of the effects of NCGs in reactor coolant system was done.

- Journal article on pool stratification and mixing induced by steam injection through spargers was published in Nuclear Engineering and Design.

- Journal article on frequency analysis of chugging condensation in pressure suppression pool system with pattern recognition was published in Nuclear Engineering and Design.

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

The objective of the project was to improve the understanding of thermal-hydraulic system behavior by performing integral and separate effects tests, 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. This data can be used in the development and validation of computer codes for the safety analyses of nuclear power plants. 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.

Finland participates in the OECD/NEA PKL Phase 4 project (2016-2020) with PWR PACTEL experiments. The OECD/NEA PKL Phase 4 project is performed with the financial support of the Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2018), the Finnish power company Teollisuuden Voima Oy (TVO), and the partners participating in the OECD/NEA PKL Phase 4 project. The authors are grateful for their support to OECD Nuclear Energy Agency (NEA), the members of the SAFIR2018 Reference Group 4 and the members of the Program Review Group and the Management Board of the OECD/NEA PKL Phase 4 project. The data from the experiments in the OECD/NEA PKL Phase 4 project will be available to the NEA member countries via their CSNI representative organizations three years after the end of the project.

Specific goals in 2018

Specific goals in 2018 were to perform a reference test for the forthcoming PWR PACTEL experiments in the OECD/NEA PKL Phase 4 project, make a TRACE simulation model of the

PKL test facility, organize a Programme Review Group and Management Board meeting of the OECD PKL Phase 4 Project in Lappeenranta, report the PWR PACTEL flow reversal due to a pump trip experiments and APROS simulations, carry out the characterizing tests of the passive heat removal test facility, and write a journal article on simulations of the PWR PACTEL nitrogen experiments.

In 2016, the experiments studying the effect of nitrogen in LOCA were carried out with the PWR PACTEL facility. The aim was independently verify the piston effect of nitrogen. In two of these experiments, the line between the upper plenum and downcomer was open and the break location in the cold leg and the accumulator injection point was close to the downcomer. Therefore, nitrogen escaped through the break and no clear piston effect was observed. In the OECD/NEA PKL Phase 4 project, new experiments are carried out with the PWR PACTEL facility where the break location is in the cold leg near the steam generator and the effect of the line between the upper plenum and downcomer to the piston effect is tested. The experiments require a reference experiment without nitrogen. The reference experiment was carried out and reported as a Quick Look report in 2018. The other two experiments will be carried out in 2019. All the experiments will be analyzed and reported with details together in 2019.

Most of the organizations participating in the OECD PKL Phase 4 Project will do analytical work with thermal-hydraulic codes. It would be beneficial for LUT also to participate in the analytical work of the experiments with the PKL facility. A TRACE simulation model of the PKL test facility was made and tested against the characterizing tests of the facility for that as a master’s thesis.

One meeting of the Programme Review Group and Management Board of the OECD PKL Phase 4 Project took place in Lappeenranta on the week of 22-25 May 2018 and was organized by LUT.

Tripping of a reactor coolant pumps causes asymmetric flow conditions in primary loops as well as in a reactor core. A flow reversal occurs in the affected loop due to the reversal of the pressure distribution in the loop caused by the other running pumps. The final flow conditions are characterized by a slight overflow in the intact loops and a backflow in the loop with the idle reactor coolant pump. In the INTEGRA project, the phenomenon was studied experimentally with the PWR PACTEL facility in 2017. The analyses and reporting of the experiment results and APROS simulations were carried out in 2018.

The SAFIR2018 INTEGRA project introduced a completely new facility, PASI, for the studies of a passive containment heat removal system. The facility was constructed in 2017 and the first experiments were performed in 2018. The reference system for the PASI test facility is the passive containment heat removal system of the AES-2006 type pressurized water reactor. This type of passive system is designed also for the planned Hanhikivi unit in Finland.

The functioning of the PASI facility is based on natural circulation. With the PASI facility, the goal is to make tests to measure system performance characteristics, and to detect issues that could disturb the operation of a passive system or prevent it from functioning as designed. The PASI test facility consists of a pressure vessel simulating containment conditions, a heat exchanger, a water pool and interconnecting riser and downcomer pipelines. Additional systems are included to provide steam, collect condensate water, remove steam and inject feed water. An aerosol injection system can be added to the system in future.

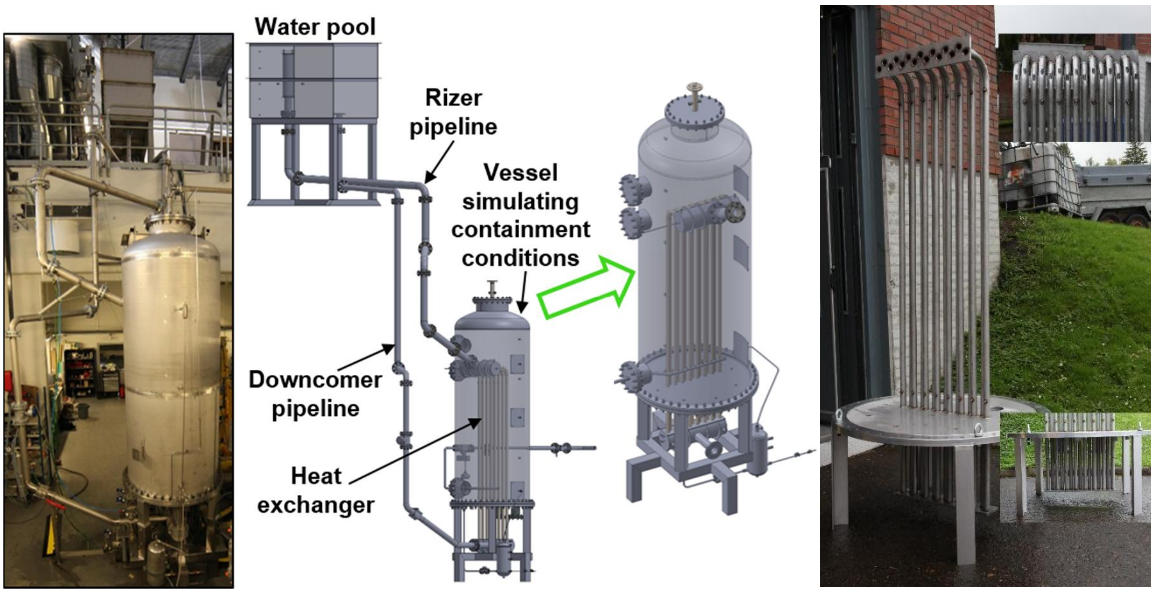


Figure 2.26. PASI test facility.

Characterizing experiments were performed to analyze, verify and study characteristics that describe this passive heat removal loop. The pressure losses were determined for normal and reverse directions. A heat up and cool down method was used to calculate integral heat losses and heat capacities of the facility over a range of fluid temperatures. The uncertainties of the heat losses and heat capacities are relatively high. There is not any single source for the uncertainties. Mostly those are rising from the approximations made for the analysis and from the uncertainties in the measurements.

A natural circulation experiment was performed to study the normal behavior of the PASI facility and the capability to remove heat. The natural circulation experiment started from atmospheric pressure and low temperature in the vessel. The PASI facility was operating as expected, transferring heat from the vessel simulating containment conditions to the water pool efficiently. The natural circulation in the loop initiated soon after the steam supply to the vessel simulating containment conditions began. The single-phase period showed steady and smooth operation of the heat removal loop, lasting about 2.5 hours. The oscillation phase started as the hot loop side reached saturation conditions and the single-phase flow turned into two-phase flow. This period was characterized by the strong oscillation of the loop mass flow rate.

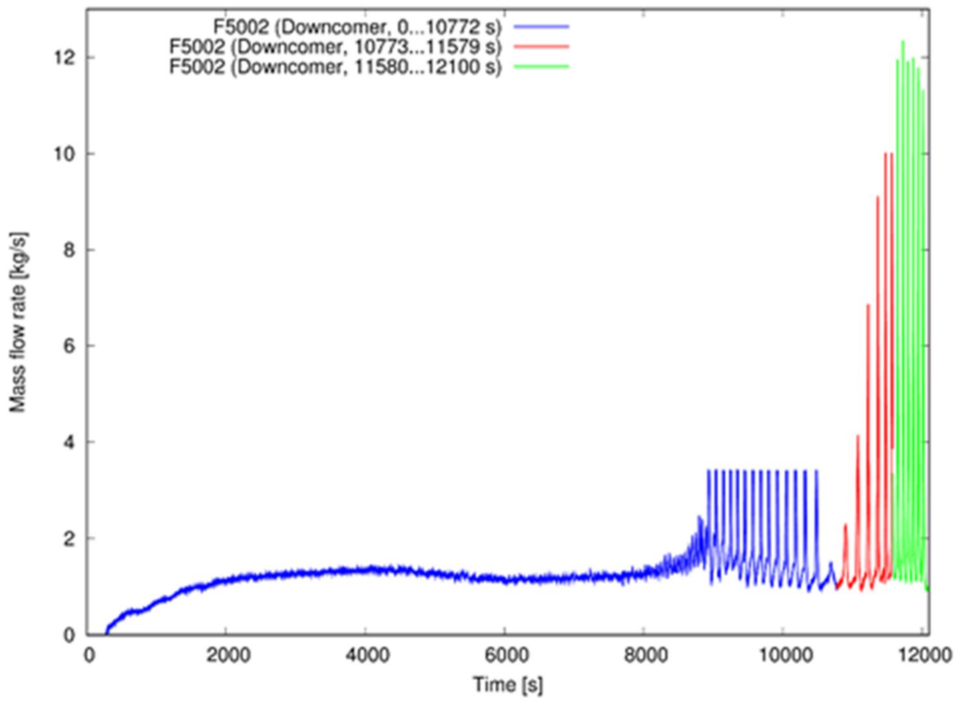


Figure 2.27. Loop mass flow rate in the natural circulation experiment. At the end of the experiment, the measuring range of the flow meter had to be changed twice because the peak values of the natural circulation mass flow rates exceeded the maximum value of the flow meter measurement range.

Deliverables in 2018

- Participating in the OECD/NEA PKL Phase 4 project with PWR PACTEL experiments
- TRACE simulation model of the PKL test facility
- Research reports of the flow reversal due to a pump trip experiments and APROS simulations
- The characterizing tests of the passive heat removal test facility
- Journal article “System code analysis of accumulator nitrogen discharge during LOCA experiment at PWR PACTEL test facility” (under review process in Nuclear Engineering and Design)

2.2.6 KATVE - Nuclear criticality and safety analyses preparedness at VTT

The general objective of the KATVE project was to maintain and develop the domestic competence in various nuclear safety analyses that may be required by the authority or the utilities. The safety analyses covered in the project were mainly related to reactor physics and radiation transport, but also heat transfer and fuel integrity analyses were included in a comprehensive safety study of a dry storage cask, which was completed during the four-year project. In practice, the KATVE project involved development and validation of calculation tools required for safety analyses, studying the domestic and international standards and requirements, and performing practical safety analyses to provide valuable experience for the research personnel.

Specific goals in 2018

One of the main objectives in the project has been the development of radiation shielding functionalities in the Serpent Monte Carlo code. The first code version supporting photon transport and thus allowing gamma shielding calculations was released in 2015. The photon interaction physics was thoroughly tested and the compared against MCNP6 with good results. The work has continued and extended to the transport physics related to severe accident management conditions when the separately proposed new RADICAL project was partly funded and merged with KATVE, effective from the beginning of 2017. The volume of the work package related to radiation transport issues remained constant for 2018.

The photon physics capabilities of Serpent were expanded by implementing a photonuclear physics model. The capability to simulate photoneutron production is needed for applications such as neutron shielding, neutron source systems, beryllium reflectors and heavy-water moderated reactors. Data-based approach was selected for simulating photonuclear reactions, which utilizes the same reaction sampling methods that are used for neutrons in Serpent, with the exception of relativistic frame-of-reference transformation and relativistic collision kinematics required for discrete two-body reactions. Photonuclear cross section data is read from an ACE format library. Because photonuclear reaction probabilities are always low (less than 5%), a variance reduction technique known as forced collisions was implemented which produces photoneutrons on every photon interaction whenever possible. Verification tests showed excellent agreement with photoneutron production yield calculated from the ENDF/B-VII.1 data.

Multiple issues in the MCNP's photoneutron physics model were detected. MCNP uses inadequate collision kinematics for discrete two-body reactions and does not perform any transformation from the center-of-momentum frame to the laboratory frame. This causes overestimation of photoneutron energies, most importantly in the case of H-2(gamma,n)H-1 reaction. Another problem is that MCNP incorrectly produces photoneutrons for many nuclides below photoneutron threshold energies, which means that MCNP overestimates photoneutron production in these nuclides. Also, MCNP doesn't take into account the important discrete two-body reactions of tungsten isotopes W-182 and W-186. As a result, MCNP significantly underestimates photoneutron production in tungsten which is often used as a radiation shielding material. Clearly, MCNP's photonuclear physics has not been verified.

Variance reduction methods are practically necessary when radiation shielding problems are calculated with any Monte Carlo code. Therefore, the implementation of such methods to Serpent has been an important part of the development of the radiation transport tools. The work in 2018 resulted in completion of a deterministic built-in importance mesh solver that applies the response matrix method to the solution of the adjoint transport problem. The new methods were utilized with CAD-based geometry input to generate a preliminary Serpent model of the hot-cell in VTT CNS.

Demonstration of the calculation chain Serpent - OpenFOAM – VTT-ENIGMA for dry storage cladding integrity assessment was performed in 2015-17, and the results were published in 2018: Arkoma, A., Huhtanen, R., Leppänen, J., Peltola, J., Pättikangas, T., Calculation chain for the analysis of spent nuclear fuel in long-term interim dry storage. Annals of Nuclear Energy, Vol. 119, pp. 129–138. That study included, i.a., cladding creep failure estimation. Another postulated cladding failure mode during dry storage is hydride induced failure, such as delayed hydride cracking. Hydrides make the cladding more brittle and that way complicate fuel handling. Hydrogen diffuses to the colder regions of the cladding, and during the drying process prior to dry storage, hydrides dissolve and again precipitate when temperature decreases in dry storage. Hydrides may re-orient and precipitate into detrimental radial hydrides, which reduce the cladding ductility. Local effects in the cladding, such as hydrogen diffusion and radial stresses cannot be handled with traditional 1.5-dimensional fuel performance codes.

In 2018, hydride effects were investigated with the fuel performance code BISON, by using the same reference case as in the demonstration in 2017. The results were typical and consistent compared to the hydrogen/hydride phenomena. Steep and relatively narrow hydride rim was formed close to the cladding outer surface when hydrogen precipitated into hydrides during dry storage. Even though the calculated cladding hoop stress in the early stage of dry storage was close to a certain hydride radial reorientation threshold, it is unlikely that the hydride content estimated with BISON is high enough to lead to a through-wall cladding failure. The complementary hydride analysis is an important addition to the dry storage simulation capabilities developed in the previous years of KATVE project, and helps to ensure the safety of long-term dry storage.

Performing valid criticality safety analyses requires that the calculation system, consisting of a transport calculation code and the cross section library, is validated for the purpose. In practice, this means modelling a large number of critical experiments with the calculation system and comparing the computational results against the experimental data to obtain an estimate for the bias of the system. To automatize the validation of the calculation codes, a validation script is being developed. The script runs a series of calculations with Monte Carlo codes Serpent and MCNP, and automatically analyses the results. The number of criticality experiments included in the validation package has increased every year.

The process took a major step forward in 2016 when VTT received a large number of MCNP inputs from the Dutch NRG and with the help of a conversion script, these could be translated to Serpent inputs rather smoothly. However, the inputs received from an external source have to be reviewed before use. This was somewhat done in 2017, when a large number of these inputs were converted to Serpent. In 2018, the work focused on review of the inputs so that they could be added to the validation package for MCNP. As a result, 95 inputs were added to the package making the total number of inputs 208. Previous work had made the number of cases in the validation package for Serpent to be 507. Additionally, comparison calculations between MCNP and Serpent were run with the experiments that are modelled for both codes. The validation script was slightly modified in 2018. The most obvious future need with the script is related to the current lack of information required for the trend analysis in the post-processing as far as MCNP inputs are considered. The parameters could be provided in the inputs - as has been done with Serpent - or in a separate file. In order to facilitate the future work with both the code inputs and the script, a version management system was established in 2018.

In order to promote the preparedness to use burnup credit in the criticality safety evaluations, Serpent calculations of the OECD/NEA burnup benchmark phase 6 were continued. The benchmark assignment was to computationally repeat the irradiation history of the VVER-440 fuel assembly, from which 8 specimens were taken and radiochemically analysed to determine the spent fuel compositions at various burnup levels. First calculations were performed and reported in 2017. These were complemented with some sensitivity calculations with respect to various burnup related modelling option and algorithms. Additionally, the impact of the used cross-section library was studied between ENDF/B-VII.1, JEFF-3.2 and JENDL-4.0.

The main objectives in the field of reactor dosimetry were to educate a new expert to handle the computational tools used at VTT and to test further the applicability of the variance reduction methods implemented to Serpent in dosimetry calculations. A spectral adjustment problem of dosimeters irradiated in the surveillance chains of Loviisa unit 1 was conducted with the LSL-M2 code. The real value of the work, however, resides in a comprehensive data processing and traceability exercise. Thorough back-tracing of all the computational steps involved in the generation of the input data files required by LSL-M2 (among which we find the fluence spectrum calculated by Serpent) from the very basic evaluated dosimetry data libraries, simulator and experimental activity data was conducted. Along this process, a vast number of computer scripts was generated in order to fill the gap between basic data and the final input files required by LSL-M2.

Among others, the work established a systematic approach to the generation of cross section and covariance data from the International Reactor and Fusion Dosimetry File IRDFF version 1.05. Along the process, a number of different approaches for the determination of the weighting spectrum is introduced, and points to the fact that the generation of the cross section library is problem-dependent. Given that the format of the XS file read by LSL-M2 is essentially the same as the one read by PREVIEW, this work also presents a hands-on guide for updating the dosimetry XS library of PREVIEW. Additionally, a damage XS library was generated following either ASTM or EURATOM standards.

Whilst attempting to trace the fluence spectrum relative covariance data, several models were tested, and a new statistical approach that is based on first principles based on data dumped by a modified version of Serpent was proposed.

This work was instrumental in preserving spectrum-adjustment-related skills, as well as in improving the quality and traceability of the new XS data, in addition to proposing a new model to account for relative correlations in multi-group fluence.

Within the field of international collaboration, several project members attended the Serpent User Group Meeting at Otaniemi, Espoo. The project also had a representative in the NEA Nuclear Science Committee meeting in June and was informed about the meeting of WPNCS.

Deliverables in 2018

- A journal article describing the calculation chain from neutronics and fuel depletion calculations to heat transfer (CFD) analysis and finally the fuel integrity analysis in a dry-storage cask was published in Annals of Nuclear Energy. The article had been written mostly in 2017.
- A journal article describing the physics models used in Serpent photon transport functionalities was submitted to Radiation Physics and Chemistry.
- A journal article manuscript documenting the variance reduction methods implemented to Serpent was submitted to Nuclear Technology.
- An extended summary - submitted to M&C 2019 conference - describes the photonuclear reaction models implemented to Serpent.
- An extended summary - submitted to M&C 2019 conference - describes the modelling of VTT CNS hot-cell with Serpent and a CAD-based geometry description.
- Report on dosimetry calculations with various VTT's computational tools to model surveillance chains irradiated at Loviisa NPP.
- Report on test calculations of the beta bremsstrahlung model implemented in Serpent. Comparison calculations were performed against MCNP and Geant4 Monte Carlo codes.
- Report on fuel cladding hydride effects in dry storage. The calculations were performed with the BISON code.
- Status report on the development of the criticality safety validation package for Serpent and MCNP in 2018. The package, consisting of code inputs and running script, can also be considered a deliverable itself.
- Report on OECD/NEA burnup credit benchmark phase VI calculation to deepen the Serpent calculations initiated in the previous year.

2.2.7 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, started in 2004, and carried out within the previous SAFIR programmes, such as the KÄÄRME project in SAFIR 2014. Compared to KÄÄRME, the work is more focused on a specific field of applications, namely spatial homogenization, i.e. the production of group constants for deterministic fuel cycle simulator and transient codes. The primary objective and expected result is a first of a kind Monte Carlo based calculation tool, capable of performing group constant generation in a routinely manner. The code can be used to complement or even replace current state-of-the-art deterministic lattice physics codes, bringing the advantages of the continuous-energy Monte Carlo method to spatial homogenization.

The methodology implemented in Serpent was largely completed and put to practice during the first two years of the project. The results demonstrated that the Monte Carlo based calculation sequence can be used for producing the full set of homogenized group constants for LWR core calculations at an acceptable computational cost. During the course of the work it was discovered, howerver, that the accuracy of the full-core calculations was limited by methodological factors. The methods used in VTT's nodal diffusion codes originate from the 1970's, and no longer represent the state-of-the-art in reactor analysis. The accuracy of legacy codes is compromised in particular near sharp material discontinuities such as the core-reflector boundary, near control rod tips, and in modern fuel types with axial profiling.

To overcome this problem, the development of a new nodal diffusion solver called “Ants” was started in 2017. The code relies on Serpent-generated group constants and modern AFEN-FENM methodology, and is capable of handling both square and hexagonal lattice geometries in steady-state and transient conditions. The work is connected to the renewal of VTT's computational system for reactor core safety analyses, which was started in mid-2017 when a new research professorship was established in the field of reactor safety. In practice, the Ants code forms the neutronics solver in a reduced-order calculation sequence of the “Kraken” computational framework.

Other specific research topics included developing methodologies for 3D homogenization to account for the effects of axial discontinuities in LWR core geometries, and including fuel temperature feedback in assembly burnup calculations, in an effort to study new approaches to state-point parametrization. The project shares topics and collaboration with the KATVE, SADE and PANCHO projects.

The international success of the Serpent code and the importance of developing a fully independent calculation system accompanied by source-code level understanding of the methodology was recognized in the SAFIR 2018 Framework Plan, where it was also recommended that the range of applications should be broadened.

Specific goals in 2018

The project was divided into three main work packages focusing on development, validation and international collaboration.

In 2017 the main focus in code development was turned from Serpent into the Ants nodal solver. The work was continued in 2018. The solver was extended from rectangular to hexagonal geometries, which allows the modelling of VVER-type reactor cores. The hexagonal capabilities were demonstrated in a conference paper presented the 28th Symposium of the

AER in October. Another major milestone was the implementation of a group constant model to cover the various state points in an operating reactor core. The Ants code is developed by two doctoral students at VTT, and the coupling of Ants into the Kraken computational framework is to be carried out in 2019 within scope of the LONKERO project in SAFIR 2022.

The Ants nodal code relies on Serpent-generated group constants. Even thought the methodology for spatial homogenization in Serpent 2 was practically completed during the first two years of the MONSOON project, several new features have been implemented to improve the methodology. Major new capabilities include improved leakage models to correct the bias resulting from infinite-lattice approximations, and more efficient algorithm for calculating the diffusion coefficient using the cumulative migration method (CMM).

Lappeenranta University of Technology (LUT) participated MONSOON in 2018 with a task to assess the performance of functional expansion tallies (FET) in fuel performance code coupling. FETs can be used in Serpent for obtaining continuous reaction rate distributions without spatial discretization. This capability is considered attractive for representing the axial power shapes of fuel pins, which using the conventional approach would neccesitate refined spatial binning. The work revealed certain challenges related to material discontinuities, which are not well represented by low-order polynomial functions. The work is continued, and a paper abstract was submitted to the M&C 2019 international conference.

The validation task suffered from major budget cuts already in 2016, and many of the original goals had to be dropped. Since the focus was later turned from VTT's legacy codes to the development of new computational tools, also the role of code validation has changed over the course of the project. The results of the Ants nodal code have been compared to various benchmark results and reference Serpent calculations, and the work continues in the LONKERO project of SAFIR 2022.

International collaboration involves support and daily interaction with Serpent users. Forms of interaction include maintaining the Serpent website, discussion forum and on-line wiki, organizing annual user group meetings and participation in the activities of the international reactor physics community. In 2018 this included participation in the ANS Reactor Physics Division meetings in June and November, and the 8th International Serpent User Group Meeting hosted by Aalto University in Finland.

Deliverables in 2018

All tasks planned for 2018 were completed, although some deliverables had to be postponed to the follow-up LONKERO project due to constraints in time and resources. Specific deliverables are listed below.

- Conference paper on the verification of the previously implemented rectangular diffusion solver in the Ants nodal code was presented at the PHYSOR 2018 conference.
- Conference paper on the estimation of the effects of the simplified fuel temperature profiles in BWR group constant generation history calculations was presented at the PHYSOR 2018 international conference.
- Conference paper on a new stochastic sub-step based burnup scheme for Serpent 2 was presented at the PHYSOR 2018 international conference.
- Extension of the nodal solver in Ants from rectangular to hexagonal geometry was presented at the 28th Symposium of AER.
- The new neutron and photon cross section libraries for Serpent 2 were documented in two research reports.

- The implementation of the first group constant model for Ants was documented in a research report.

- Serpent-TRANSURANUS coupling and performance tests for function expansion power tallies were described in a research report, and a paper abstract was submitted to the M&C 2019 international conference.

- A conference paper abstract on the development of the Kraken computational framework was submitted to the M&C 2019 international conference.

- The international Serpent user community grew from 760 users in January 2018 to 900 users by the end of January 2019. The code has users in 217 universities and research organizations in 42 countries around the world. The Serpent website lists some 750 peer-reviewed scientific journal articles and conference papers and 175 theses published on Serpent-related topics worldwide.

- One source code update (2.1.30) wase distributed to Serpent users in 2018.

- The 8th Annual Serpent User Group Meeting was hosted by Aalto University on May 29 - June 1, 2018.

- A Serpent workshop was organized at the PHYSOR 2018 international conference.

- International collaboration also included participation in the Executive Committee meetings of the Reactor Physics Division of American Nuclear Society in June and November 2018.

- Writing of a User Manual for Serpent 2 as an on-line Wiki was continued in 2018.

- Several papers from Serpent user organizations were co-authored in 2018.

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

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.

The NURESA project consisted of four Work Packages (WP), where Computational Fluid Dynamics (CFD) methods were developed and validated for nuclear reactor safety assessment.

In WP 1, the international blind benchmark in the Hydrogen Mitigation Experiments for REactor Safety (HYMERES) programme was participated. The benchmark exercise focussed on the hydrogen stratification and erosion of density layer by turbulent mixing processes.

In WP 2, spray and stratification experiments performed with the PPOOLEX facility at LUT was modelled with CFD simulations. In addition, separate effect condensation experiment performed with the SEFPOOL facility at LUT was calculated with ANSYS Fluent and OpenFOAM codes. In the experiment, horizontal steam jets are injected into water pool, where direct-contact condensation of steam occurs.

In WP 3, OpenFOAM CFD solver was developed and validated for nuclear reactor safety assessment. At VTT, subcooled nucleate boiling and wall heat transfer models were integrated into the Eulerian two-phase solvers of the official OpenFOAM release. In addition, size distribution model of bubbles was implemented in co-operation with Helmholtz-Zentrum Dresden-Rossendorf (HZDR). At LUT, OpenFOAM simulations of POOLEX and PPOOLEX chugging experiments were done and models for direct-contact condensation were developed.

In WP 4, coupled simulation of VVER-440 steam generator with ANSYS Fluent and Apros 6 has been performed. In addition, coupled simulation of VVER-440 pressurizer was performed, where CFD model of the pressurizer was coupled with generic model of VVER-440 nuclear power plant.

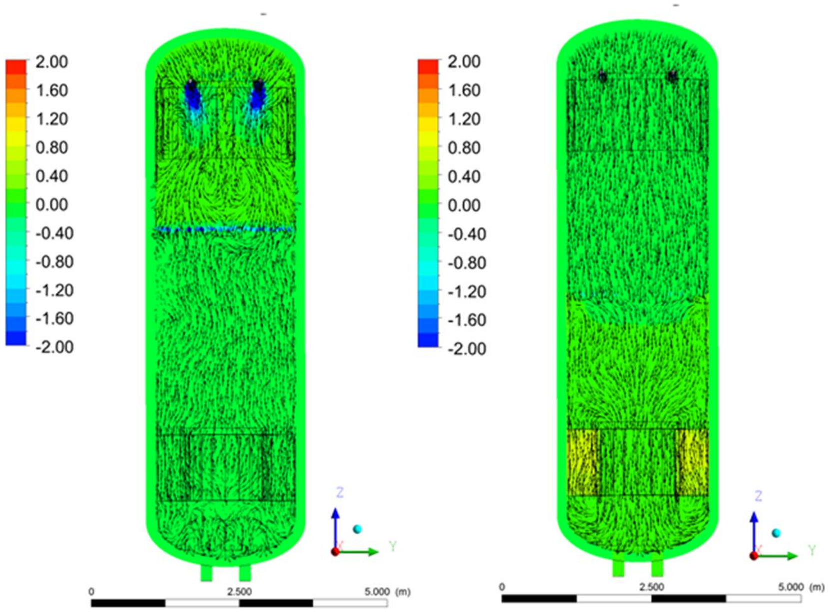


Figure 2.28. Mass transfer rate [kg/m^3·s] and velocity vectors during pressure decrease (left) and pressure increase (right). Negative value is condensation and positive evaporization. Values below -2 are coloured blue and values above 2 red.

2.2.9 PANCHO - Physics and chemistry of nuclear fuel

Nuclear fuel both produces the energy in nuclear power plants and acts as the first two barriers to the spread of radioactive fission products. The UO_2 matrix of the fuel pellets contains approximately 99% of the born radionuclides, while the cladding tube contains the rest. Therefore the integrity of the fuel during normal operation and accidents is of utmost importance. Traditionally fuel performance has been analysed with integral fuel codes that contain semi-empirical correlations deduced from experiments. These correlations and models become more and more mechanistic as the understanding and the demands increase.

The project PANCHO – Physics and Chemistry of nuclear fuels investigates the integral fuel behaviour as well as combines the experimental and the modelling approaches in studying several topical features of nuclear fuel behaviour. These topics are the chemistry of the fuel pellet and the mechanical response of the cladding.

Specific goals in 2018

Specific goals in 2018 in the PANCHO project included the validation and further development of FINIX, participation in the OECD/NEA RIA Fuel Codes Benchmark, application of the previously developed FRAPTRAN-GENFLO coupling to model blocked fuel bundles and modelling missing pellet surface defects with the three-dimensional fuel performance code BISON.

The final version published in the PANCHO project is 1.19.1 (Loukusa et al., 2019), which includes a cladding oxidation model for Zircaloy-4 and a rewritten coolant heat transfer model for standalone purposes. A better coolant model was decided to be implemented to yield more accurate cladding oxidation pre-dictions, as FINIX could not previously model, for example, the axial rise in cool-ant temperature. A Master’s thesis is currently being finalized based on the clad-ding oxidation and coolant models in FINIX. The latest version of FINIX predicts fuel temperatures within 6.6 % of experimentally measured values (Peltonen 2019), which is comparable to other fuel performance codes. The temperature validation results are shown in upper figure in figure 2.29.

In addition, FINIX-1.19.1 was used to model loss-of-coolant accidents for the first time. Even though the cladding mechanical models are insufficient for large strain deformation modelling, interesting and promising results to guide future development were obtained. Especially the inclusion of finite strain modelling, improvements in rod failure predictions and an update of the plenum tempera-ture model were thought to be of most importance. In some cases, very good agreement with FRAPTRAN was obtained, such as in the fuel temperatures and rod internal pressure. The rod internal pressure predictions of FINIX are com-pared with FRAPTRAN in the lower figure in figure 2.29.

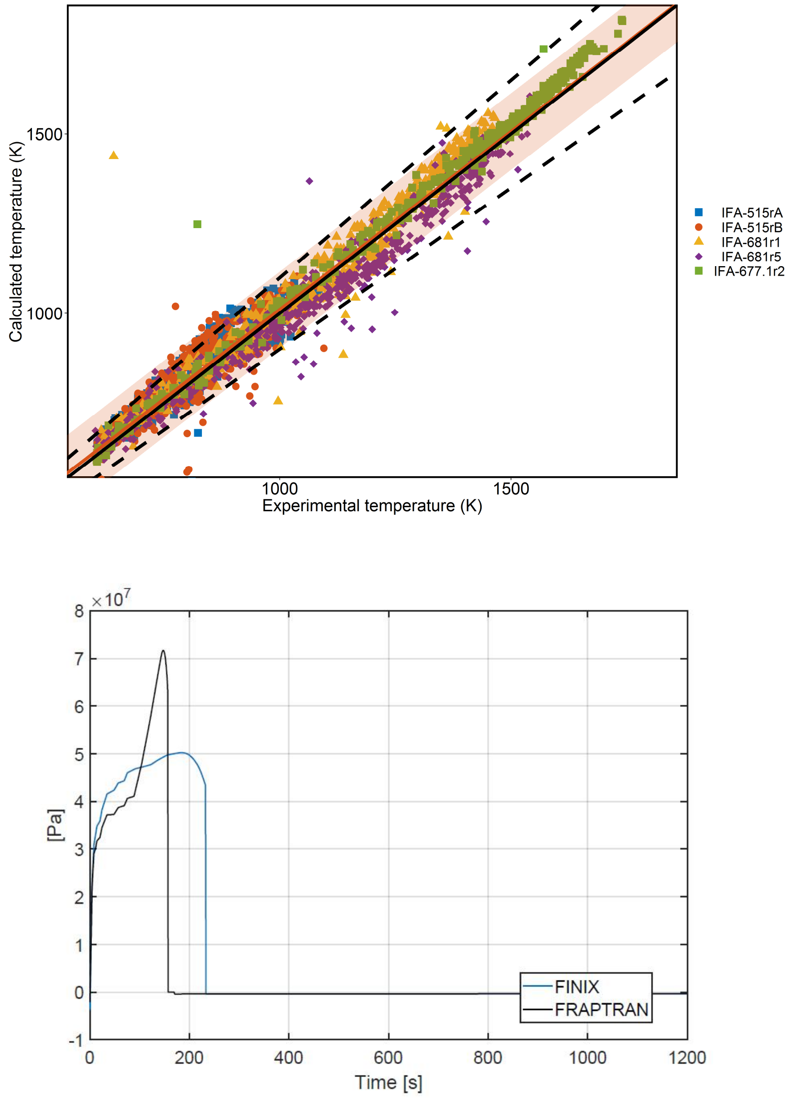


Figure 2.29. Temperature validation results of FINIX in selected Halden irradiations (upper) and .rod internal pressure predictions of FINIX in the IFA-650.5 LOCA test compared to FRAPTRAN (lower).

As a follow-up of the OECD/NEA RIA fuel codes benchmark that VTT participated , Phase 2 of the benchmark was organized. It was divided into two parts: in Activity 1 (years 2014-2015), ten simplified (fabricated) RIA cases were simulated in order to understand the differences in modelling in various codes. In Activity 2 (2015-2016), uncertainty and sensitivity analyses were done for a selected fresh fuel (but no pellet-cladding gap) PWR case from Activity 1. The number of uncertainty runs per participant was 200. Various influential input parameters were identified by using partial rank correlation coefficients. The results of Phase 2 cannot be extrapolated to cover irradiated fuel, and therefore, in Phase 3, a real RIA test with irradiated fuel was analysed with the methodologies agreed in Phase 2. Due to the large discrepancies regarding RIA thermal hydraulics modelling found in Phase 1 and 2, the uncertainty simulations considered CABRI International Programme (CIP) test CIP0-1 performed in a sodium loop. An example of sensitivity study results obtained at VTT is shown in Table 2.4. A synthesis of participants' results and the conclusions of Phase 3 will be published in an NEA report in 2019.

Table 2.4. An example of sensitivity study conducted in RIA benchmark Phase 3 by calculating the partial rank correlation coefficients at the time of maximum power pulse. Blue colour indicates coefficients with absolute value at or above 0.25, and red colour at or above 0.75 (range being from zero, i.e., no significance, to one).

	Output				
	Thermal behaviour		Mechanical behaviour		
Input parameter	Fuel	Cladding	Fuel	Cladding	Fission gas release
Pellet-cladding radial gap					
Cladding roughness					
Fuel roughness					
Cladding outer oxide thickness					
Injected energy in the rod					
Radial power profile					
Power pulse width					
Fuel thermal conductivity model					
Fuel thermal expansion model					
Fuel enthalpy					
Cladding thermal expansion					
Cladding yield stress					

Vacancies, starting from single missing atoms with low concentrations, specify many of the important physical properties of the oxide ceramics, such as ther-mal conductivity and gas diffusion coefficients. In nuclear fuel, thermal conductivity is closely related to the reactor safety and the energy conversion efficiency. Volatile element diffusion, in turn, ulti-mately determines fission gas release in the fuel rod, which is an important safety aspect affecting the rod design.

In some cases, it is reasonable to overcome the challenges caused by radio-activity and licensing of the samples via study and comparison of a non-radioactive surrogate material for nuclear fuel. CeO_2 is a widely used non-radioactive surrogate material for UO_2 and PuO_2 having the same fluorite struc-ture.

The work in PANCHO in 2018 aims to characterize vacancy-type defect evolution in proton-irradiated CeO_2 at 600 °C and connect it to the physical properties of the sample, such as thermal conductivity. So far the project includes fabrication of the sam-ples, irradiation of the samples, and X-ray diffraction characterization at SCK-CEN Belgium, positron annihilation characterization and DFT calculations at Aalto University, and thermal conductivity measurements at Ohio State University, Columbus, United States, the latter studies being ongoing. Work in this subject performed in PANCHO has concentrated on research coordination and scientific article preparation.

During the PANCHO project, the three-dimensional fuel performance code BISON developed at the Idaho National Laboratory (US) was taken into use at VTT. The investigations at VTT centered on general use of the code. A three-dimensional fuel performance code has not previously been in use at VTT, and several procedures in applying such codes had to be developed. The main challenge is the generation of computational meshes when the user wishes to calculate nonstandard geometries.

The workflow in setting up BISON simulations and the generation of meshes has been refined at VTT, and the BISON code has been successfully installed to the VTT Linux cluster Potku2. This preparatory work eases future simulations of advanced fuels and claddings where either the geometry of the fuel is complex or several material layers are present, such as in coated claddings.

The international cooperation such as fuel behaviour part of VTT - Halden Reactor Project in-kind work, participation in working groups OECD/CSNI WGFS and ETSON SAG, as well as the following of CABRI International Project progress is done under this project.

Deliverables in 2018

- FINIX was validated comprehensively for the first time since 2013, and the associated validation report was produced. Two versions of FINIX were packaged during 2018, and a validation report was produced for both.

- A new version of FINIX was published along with the associated user's manual and code description. The new version contains the cladding oxidation model and the previous coolant model of FINIX was updated and can now model PWR conditions. Another user's manual was produced after the first validation, where only error corrections and small changes to the code were made.

- The OECD/NEA RIA Fuel Codes Benchmark Phase III was participated in. The activity in 2018 consisted of a sensitivity analysis on irradiated fuel, where the CIP0-1 test performed in the sodium loop of the CABRI reactor was used as the test case.

- A conference article on the application of constrained Gibbs energy minimization to nuclear fuel thermochemistry was prepared and published in TopFuel 2018. Whereas typically Gibbs energy minimization can only be used to calculate the equilibrium composition of a substance, with constrained Gibbs energy minimization, nonequilibrium states can also be calculated. The method was applied to the radiolysis of cesium iodide in the pellet-cladding gap.

- The BISON fuel performance code was used to model a missing pellet surface defect. For the first time at VTT, self-made meshes for BISON were generated and used successfully with BISON. The development allows for more detailed future 3D modelling of nuclear fuel at VTT.

- Previously made measurements were collected into a journal article draft on proton-irradiated cerium oxide and the determination of vacancy cluster behavior with temperature.

- A journal article draft on using the previously developed FRAPTRAN-GENFLO coupling was unfinished at the end of the project due to difficulties in the mechanical modeling in FRAPTRAN. The work will be continued in the future to prepare the journal article.

2.2.10 SADE - Safety analyses for dynamical events

Aim of the SADE project has been to develop the modelling of transient events and accidents such that we can give more reliable answers to the safety requirements set in the YVL guides. To achieve this, the VTT's capabilities for independent transient safety analyses has been improved by coupled use of the CFD-type thermal-hydraulics solver PORFLO and the reactor dynamics codes HEXTRAN and TRAB3D. In addition, the neutronics modelling needs to be more detailed to get the full benefit on this improved accuracy of the thermal-hydraulics modelling. The goal is to have at VTT a fully self-developed, independent calculation system that can be used for the whole calculation sequence from basic nuclear data to coupled 3D transient analyses. Aim is the tool, which is more accurate and still fast and robust enough for practical safety analysis The developed computational tool set of coupled neutronics, system codes and true-3D thermal hydraulics has been be tested and demonstrated in cases relevant from safety analyses point of view. Objective has been that by the end of the project we have calculated several transients and accidents of real interest. Developing and maintaining our own codes and in-depth understanding of them enables the best possible expertise on safety analyses.

Specific goals in 2018

The project has two main research areas. The objective of the first work package is to enhance the core modelling of VTT's 3D reactor dynamics codes. In 2018, aim was to finalize and test implementation of axially heterogeneous fuel models and axial discontinuity factors in HEXTRAN so that it can more reliably model transients of reactor cores with modern fuel assemblies. Also modelling of fuel behaviour is enhanced in this work package. In 2018 aim was to update coupling between the reactor dynamics codes and fuel behaviour module FINIX so that up-to-date version of the FINIX can be used for reactor dynamical simulations, and enabling reactivity coefficient modifications and sensitivity studies with HEXTRAN-FINIX code package.

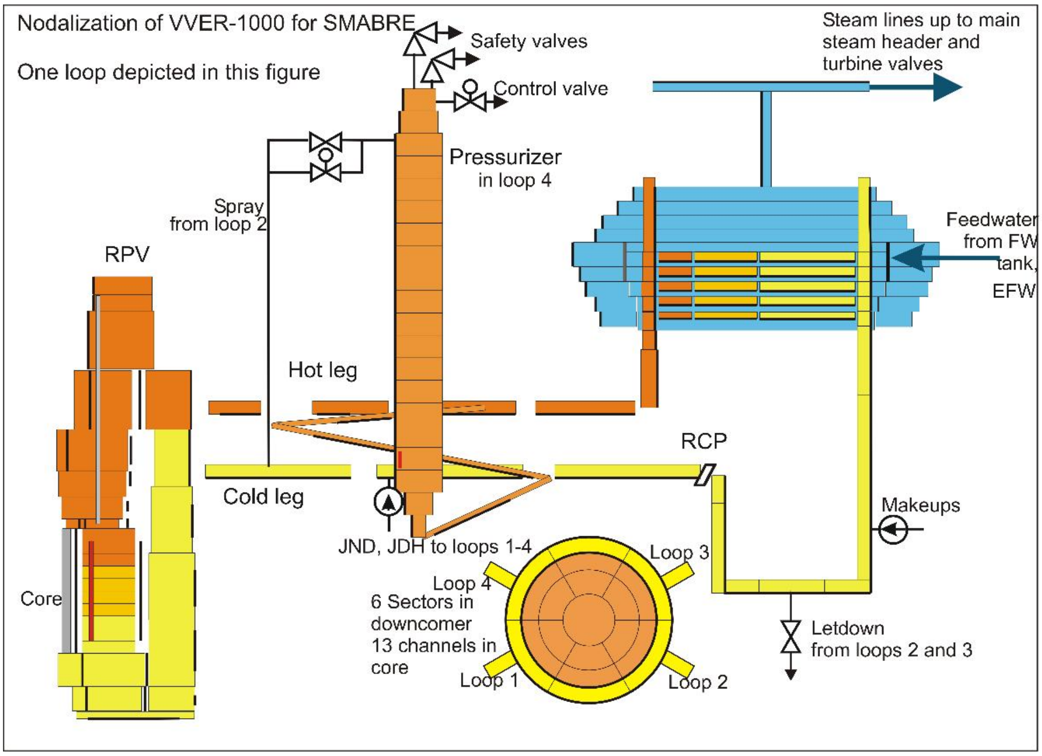


Figure 2.30: Overview of the primary side nodalization of Kalinin-3 VVER-1000 for SMABRE.

The second work package focuses on whole core transient analyses, focusing on cases where mixing in reactor pressure vessel or open core geometry play an essential role. Aim has been to further develop tools that enable more realistic modelling of the transients, and to simulate transients with these improved tools. Modelling and development has had two parallel branches: development of the tools such as internally coupled HEXTRAN-SMABRE that can be routinely used for safety analyses already during the SAFIR2018 program, and modelling of transients with CFD-style codes that have more detailed description. In 2018 aim was to validate internally coupled HEXTRAN-SMABRE for VVER-1000 reactors. One validation case is OECD/NEA Kalinin-3 benchmark problem, switching-off of one of the four operating main circulation pumps at nominal reactor power. For that case aim was to build HEXTRAN-SMABRE model for Kalinin-3 NPP on the base of existing VVER-1000 reactor models. In addition, application and validation of the fully coupled HEXTRAN-CFD-SMABRE reactor analysis code package has been continued. In 2018, aim was to perform three whole-plant simulations with the new simulation tool and at least in one case use OpenFOAM instead of PORFLO as a CFD solver.

The third work package involves work that supports the project's research aims and promotes the usefulness of the code system. The work package includes international co-operation and administration work demanded by SAFIR2018 program.

Deliverables in 2018

- Calculation model for axially heterogeneous fuel has been implemented to HEXTRAN. The new input routines have been tested, and the verification process of the modified HEXTRAN code is nearly complete. The new input syntax and changes to the source code have been reported. The verification process of the modified HEXTRAN continues. Work has been reported in a research report.
- FINIX-0.13.9 has been replaced with structurally considerably different FINIX-0.17.12 in HEXTRAN. Preliminary testing has been completed but validation of the coupling will have to be carried out later. Implementation of the renewed coupling has been described in a research report.
- HEXTRAN-SMABRE was renewed and supplemented with new, internally coupled simulation mode in 2015. In 2018, several modifications have been done to this new code version. In 2018 validation of the new HEXTRAN-SMABRE version including internal simulation mode has been continued with VVER-1000 cases. In 2018, two main validation cases for VVER-1000 reactors have been OECD/NEA/NSC coolant transient benchmark V1000CT-2 concerning hypothetical main steam line break (MSLB) at Kozloduy-5 plant and OECD/NEA/NSC Kalinin-3 benchmark problem, switching-off of one of the four operating main circulation pumps at nominal reactor power. Program modifications as well as validation simulations have been documented in a research report.

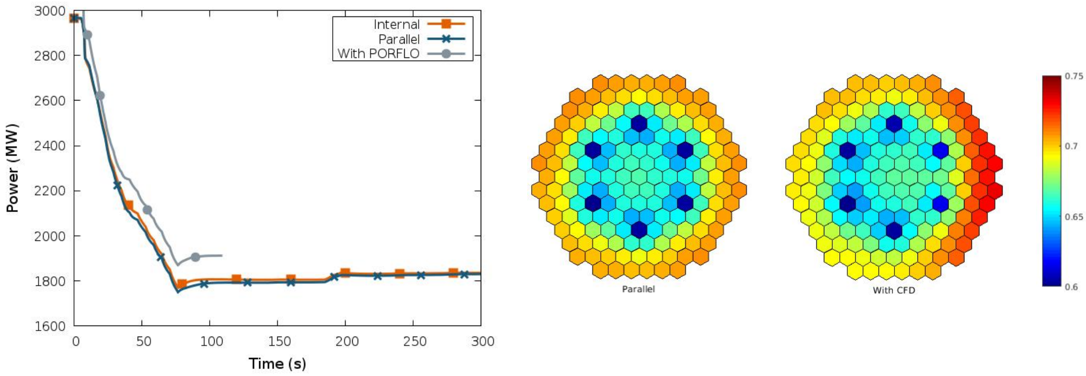


Figure 2.31: Fission power during Kalinin-3 benchmark transient (left) and assemblywise power 50 s after switching of one main coolant pump of working four RCPs versus power at initial state with parallelly coupled HEXTRAN-SMABRE and with HEXTRAN-SMABRE-PORFLO.

- HEXTRAN-SMABRE model for Kalinin-3 VVER-1000 NPP has been created on the base of the existing VVER-1000 reactor models. OECD/NEA/NSC Kalinin-3 benchmark problem, switching-off of one of the four operating main circulation pumps at nominal reactor power, has been simulated using both parallelly and internally coupled simulation mode of the HEXTRAN-SMABRE code. Model report has been completed.

- HEXTRAN-OpenFOAM and OpenFOAM-SMABRE interface routines have been prepared so that OpenFOAM can be used together HEXTRAN and SMABRE instead of the PORFLO code.
- Several transients and accidents have been simulated using fully two-way coupling between HEXTRAN, SMABRE and PORFLO:
  - OECD/NEA dynamic benchmark V1000CT-2 - scenario 1 VVER-1000 main steam line break (MSLB)
  - OECD/NEA dynamic benchmark V1000CT-2 - scenario 2 VVER-1000 main steam line break (MSLB)
  - OECD/NEA Kalinin-3 MCP benchmark

  Scenario specific inputs have been prepared for all these cases. Interface routines have been completed for reversed flows. Results have been compared to with those of HEXTRAN-SMABRE.
- OpenFOAM input files were prepared for VVER-1000 pressure vessel. A transient simulation of the OECD/NEA dynamic benchmark V1000CT-2 - scenario was computed with the new analysis framework HEXTRAN-OpenFOAM-SMABRE. Results were compared with other modelling approaches.
- The project included also participation in the AER working group D meeting in Dresden, Germany, where the presentation was given on development of a VTT's new nodal code ANTS. Also kick-off meeting of OECD/NEA Rostov-2 benchmark in Lucca Italy was participated.

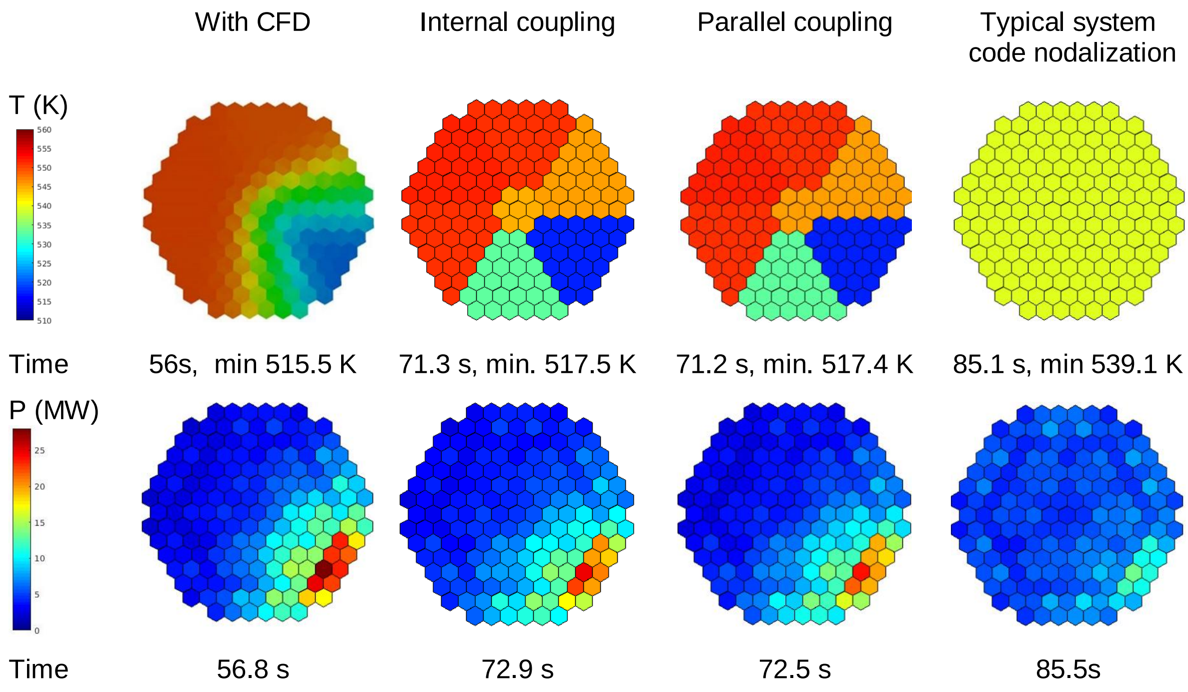


Figure 2.32. Core inlet temperature at time of minimum temperature, and assembly-wise fission power at time of maximum fission power during the main steam line break of VVER-1000 reactor with distinct coupling methods between HEXTRAN, SMABRE and PORFLO. OECD/NEA V1000CT-2 benchmark.

2.2.11 USVA - Uncertainty and sensitivity analyses for reactor safety

The general goal of the USVA project was to develop methods and practices in uncertainty and sensitivity analyses of multiphysics problems and calculation sequences in reactor safety. The goal supports the long-term aim of establishing a comprehensive methodology for uncertainty and sensitivity analysis for the entire reactor safety field. The project built on the existing expertise in uncertainty and sensitivity analysis at VTT and Aalto University, and gathered the on-going research activities under one project. New experts in this area were also trained. USVA promoted activities at the interfaces of the different disciplines in reactor safety.

Many of the tasks in USVA were related to the topics of the OECD Nuclear Energy Agency (NEA) Benchmark for Uncertainty Analysis in Modelling (UAM) for the Design, Operation and Safety Analysis of LWRs.

Two publications made during the previous years of USVA project were used in an article based doctoral dissertation defended in 2018 (Arkoma, Modelling design basis accidents LOCA and RIA from the perspective of single fuel rods, Aalto University).

Specific goals in 2018

WP1, methods and analyses, had three subtasks in 2018.

Task 1.1 continued the development of sensitivity analysis methodology for statistical fuel failure simulations of EPR large break loss-of-coolant accident (LB-LOCA) started in 2015. In 2016, it was demonstrated that support vector machines (SVMs) can be used as a surrogate model to replace some of the computationally expensive rod-level FRAPTRAN-GENFLO simulations in estimating the number of failed fuel rods in one global accident scenario. Applicability of SVMs to sensitivity analysis was demonstrated in 2017. In 2018, the work was extended to the full array of global simulation data and its sensitivity analysis. Sensitivity indices between global parameters and cladding maximum hoop stress were calculated, and that way the importance of global input parameters was studied. The global parameters that were varied in the original statistical analysis included input parameters of the system code APROS, and model parameters of the steady-state fuel behaviour code FRAPCON and the subchannel thermal hydraulics code GENFLO. The results showed that the global parameters had actually negligible effect compared to the local parameters on cladding maximum hoop stress in LB-LOCA.

Task 1.2 was a three-year task that started in 2016 with a literature review of potential methods for determining input uncertainties of thermal hydraulic codes. Due to personnel changes, a new expert on the subject was trained in 2018. The focus was on drawing up an updated plan on the application of the best-suited methodologies to the quantification of uncertainties related to physical models of APROS. Methods were reviewed and a path forward suggested.

Task 1.3 utilized the sensitivity/perturbation calculation capability implemented in Serpent 2.1.29 in USVA in 2017. In 2018, Serpent was further extended to enable reading, processing and interacting with the multi-group nuclear covariance data. This data was applied with the above mentioned sensitivity calculation capabilities to propagate nuclear data uncertainties into several different group constants that can be expressed as reaction rate or detector tally value ratios. The capabilities were demonstrated in calculations where detectors were set up to tally reaction rates, and fluxes and the group constants were calculated as a post processing step. As an example, Fig. 2.33 shows the contributions to the total uncertainty of the homogenised one group fission cross section in a hot-full-power pin cell for the Three Mile Island (TMI) 1 fuel type from UAM. In the future, a sampling based method should be used to verify the propagated nuclear data uncertainties.

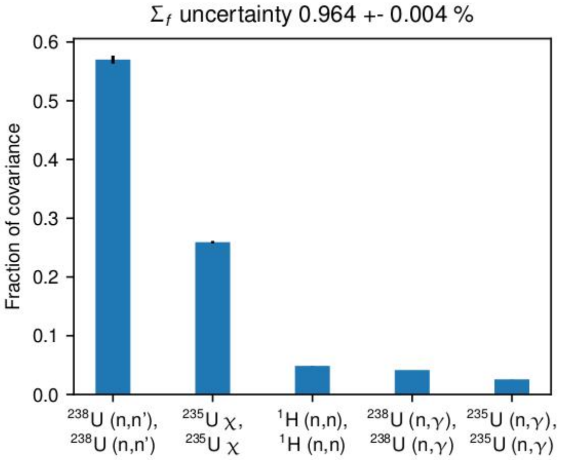


Figure 2.33. Top contributors (as a fraction of the total covariance) to the uncertainty of the macroscopic fission cross section homogenized by Serpent in a TMI1 (PWR) pin-cell.

WP2 was dedicated for uncertainty and sensitivity analyses of reactor dynamics codes, and it had one subtask in 2018. The objective was to model UAM benchmark Phase 3 problems, with the primary area of interest being the VVER-1000 task. The final specifications for this phase have not been finalized but the group constant data containing propagated uncertainties has already been prepared for the VVER case. Preliminary simulations with this data were done in USVA in 2018. For this work, the HEXTRAN-SMABRE model of Kalinin-3 VVER-1000 plant was utilized, and 500 simulation runs of the benchmark problem, switching off one main coolant pump (MCP) of working four MCPs, were completed (Fig. 2.34). Additionally, the effect of nodalization of the pressure vessel on coolant mixing was studied.

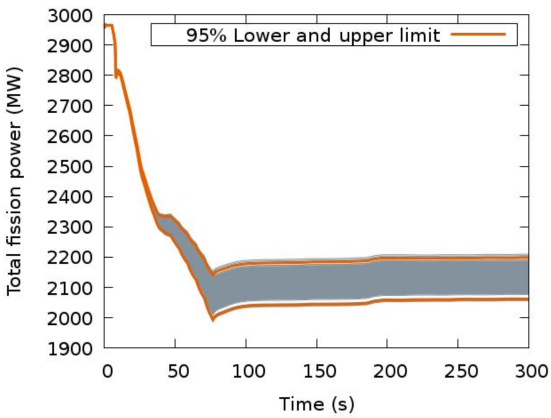


Figure 2.34. HEXTRAN-SMABRE simulation of Kalinin-3 benchmark, switching off one main coolant pump (MCP) of working four MCPs using 500 separate group constant sets.

Deliverables in 2018

- Research report on large break loss-of-coolant accident sensitivity analysis related to uncertainties of global parameters.

- Research report containing a literature review of potential methods for determining input uncertainties in thermal hydraulic codes.

- Research report on uncertainty propagation capabilities augmented to Serpent Monte Carlo code.

- Research report on preliminary uncertainty simulations performed for the OECD/NEA UAM benchmark on VVER-1000 plant with HEXTRAN-SMABRE code.

2.3 Structural safety and materials

In 2018 the research area “Structural safety and materials” consisted of nine projects:

1. Experimental and numerical methods for external event assessment improving safety (ERNEST)
2. Fire risk evaluation and Defence-in-Depth (FIRED)
3. Analysis of fatigue and other cumulative ageing to extend lifetime (FOUND)
4. Long term operation aspects of structural integrity (LOST)
5. Mitigation of cracking through advanced water chemistry (MOCCA)
6. Thermal ageing and EAC research for plant life management (THELMA)
7. Non-destructive examination of NPP primary circuit components and concrete infrastructure (WANDA)
8. Condition monitoring, thermal and radiation degradation of polymers inside NPP containments (COMRADE)
9. Evolving the Fennoscandian GMPEs (EVOGY).

2.3.1 ERNEST - Experimental and numerical methods for external event assessment improving safety

The general objective of ERNEST (2016-2018) is to develop and take into use improved methods and modelling techniques for aircraft impact analysis which are validated against experimental results. Wealth of experimental data on nonlinear dynamic behaviour of reinforced concrete structures loaded by hard and deformable projectiles has been obtained at VTT previously within IMPACT projects (Phases 1, 2 and 3). Some additional tests tailored to domestic purposes will be carried out within the ERNEST project. Models and methods for assessing structural integrity of impact loaded reinforced concrete structures are developed and validated utilising this experimental data. Knowledge transfer, training and education of new experts is carried out within this kind of working process. The results will be reported mainly as scientific and conference papers.

Specific goals in 2018

WP1 of ERNEST concentrates on testing of reinforced concrete structures under impact loading. In 2017, a double-slab structure was tested against impact by a hard projectile. In 2018, the goal was to study the effect of maximum grain size of concrete on the punching resistance of a plain concrete slab.

WP 2 of ERNEST focuses on development and validation of computational tools and models that can be used for analysing a real aircraft crash. In 2016, development of a new concrete material model was started within the project. In 2017 some very encouraging preliminary results were obtained and presented in SMiRT-24 conference. The goal for 2018 was to further refine the material model and to validate it, step by step, against different kind of impact test cases. The first cases to be taken into consideration were to be hard missile punching experiments (for example A-series of Impact projects and IRIS benchmark P-series) but gradually, the focus was to be shifted towards combined punching and bending tests (for example X-series from Impact projects)
.

Deliverables in 2016

- A research report describing the two punching resistance tests of plain concrete slabs with different maximum aggregate sizes.

- A journal paper submitted to Rakenteiden mekaniikka (Journal of Structural Mechanics) about usage of Abaqus concrete damaged plasticity (CDP) material model in simple stress states

- A journal paper submitted to Rakenteiden mekaniikka (Journal of Structural Mechanics) about usage of Abaqus concrete damaged plasticity (CDP) material model in impact simulations

2.3.2 FIRED - Fire risk evaluation and Defence-in-Depth

The main objective of the FIRED –project is to develop the tools for fire risk evaluation and create a new methodology for assessing the defense-in-depth in the context of fire safety. In general, the results may be divided into three categories: the First one is the basic research that increases understanding and contributes to future work, second is the education of experts and developing the current methodology, and the third one is the direct applications to NPPs. The results of FIRED work packages are illustrated in Figure 2.35.

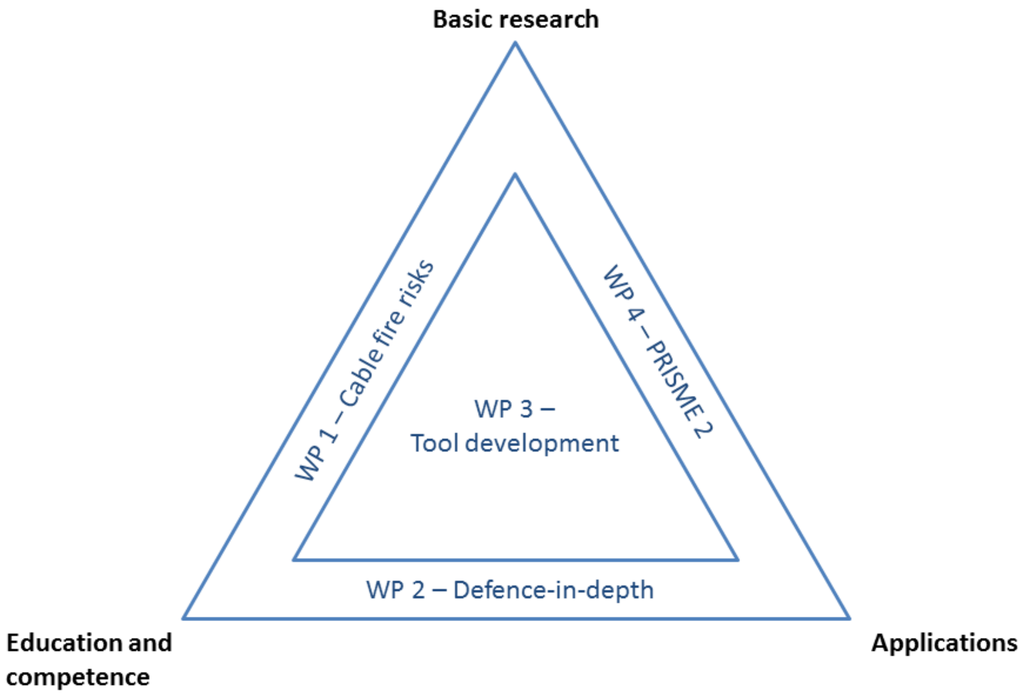


Figure 2.35 Result categories in WPs of FIRED.

Specific goals in 2018

The active tasks during 2018 were:

- WP 1: Cable fire risks during plant life cycle
  - Task 1: New flame retardant polymers,
- WP 2: Fire-Barrier performance assessment
  - Task 1: Barrier performance assessment with Fire-CFD.
- WP 3: Fire simulation development, maintenance and validation.
  - Task 3: Pyroplot development, maintenance and validation.

- WP 4: Participation to PRISME3.
    - Task 1: participation fee.

In WP1, we carried out reactive molecular dynamics simulations based on the ReaxFF reactive force field to study the effect of aluminium (tri)hydroxide on the thermal decomposition of polyethylene. The simulations reproduced the endothermic decomposition of aluminium (tri)hydroxide into alumina and water. Other known mechanisms of flame retardancy, such as heat absorption by the filler and its residue, were reproduced with reasonable accuracy. The simulations also revealed a chemical interaction between polyethylene and aluminium (tri)hydroxide, in which hydroxyl radicals released by the aluminium (tri)hydroxide abstracted hydrogen from the surrounding polyethylene, resulting in enhanced water production and enhanced charring of polyethylene.

Furthermore, we analysed the data in order to obtain kinetic parameters for the pyrolysis model of the Fire Dynamics Simulator. Specifically, a set of thermal decomposition simulations using linear heating rates in the range 0.3 - 5 K/ps were performed for model systems of pure aluminium trihydroxide (modelled as gibbsite), pure amorphous polyethylene, and a composite of PE+ATH. The simulations can be regarded as a set of atomistic scale thermogravimetric experiments. Kinetic parameters for the decomposition reaction of each system were extracted from the data using two model-free isoconversional methods, where 'model-free' means that no assumptions need to be made on the reaction model. It was enough to assume an Arrhenius-type behaviour for the reaction rate constant, and to derive the associated pre-exponential factor A and activation energy E_a from the isoconversional analysis. Examples of results are shown in Figure 2.36 for pure polyethylene.

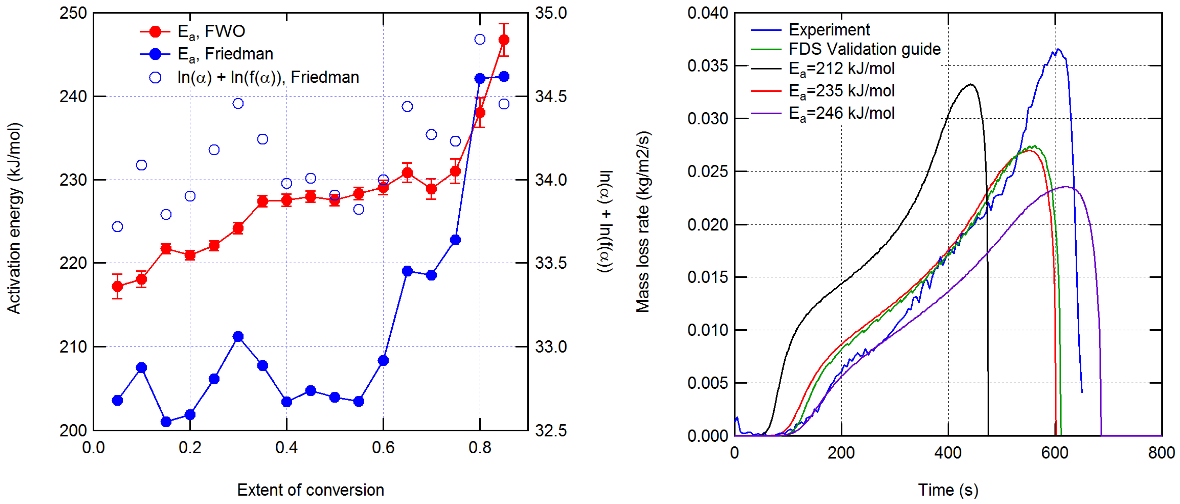


Figure 2.36 Left: kinetic parameters for polyethylene decomposition from isoconversional analysis of RMD data. Right: FDS simulation of the cone calorimeter experiment under nitrogen atmosphere using the kinetic parameters form RMD.

In WP2, work aimed at extending the previous study carried out to develop a 1D numerical model capable of predicting the fire resistance of stone-wool based fire barrier. Using the previously developed 1D model, we simulated the fire resistance test for three different types of stone-wools. We also presented separate 1D and 3D models where the standard fire was alternatively modeled using Neumann boundary condition. For model validation, small-scale fire resistance test data was collected on samples measuring 60 x 60 cm and consisting of a 1 mm thick steel sheet and three different types of wool layers (thickness between 60 and 76 mm). Figure 2.37 shows the predicted and measured cold-side temperatures for the three wools and the three different models used.

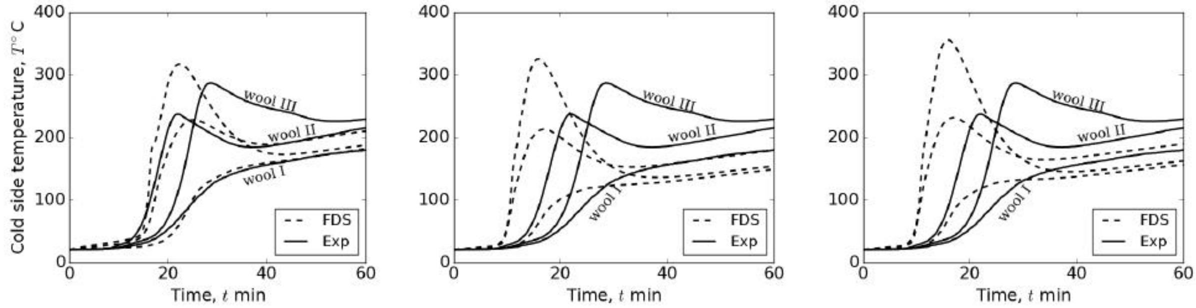


Figure 2.37. Measured and predicted cold side temperature. Left: 1D model. Middle: Alternative 1D model. Right: 3D model.

For the 1D model, the left plot shows that the predicted values are close to the measured one for wool I and II but not for wool III. When the stone-wool is heated, the binder and oil undergo exothermic reactions and provide additional heat to the stone-wool fibers. In the case of wool I, the binder content is relatively low and the heat released due to the exothermic behavior is not sufficient to influence the cold-side temperature. For wool II and III, the binder content is sufficiently high and the additional heat release due to the exothermic reaction affects the cold-side temperature, i.e., the asymmetric bell shape observed at 20 min and 25 min for wool II and III respectively. The density of wool II, however, is much lower than that of wool III. Predicted values being correct for wool II and incorrect for III, suggests that the heating of the test sample having both the high density and high oil or binder content cannot be correctly simulated using the current model.

The middle and right plots show that the alternative 1D model and 3D model prediction are matching to each other but not to the measured values. The rise in the predicted cold side temperature at the beginning of the test is much quicker than the measured one. The possible reason could be the incorrect heat flux representing the standard fire. The heat flux specified on the face of the test sample depends only upon the standard fire curve. While in reality it also depends upon the hot side surface temperature of the test sample. As the hot side surface temperature of the test sample begins to rise the heat flux growth rate should decrease. This effect is not properly modeled in the current setup.

In WP3, a new tool called PyroPython was developed as a successor to the Pyroplot tool devloped in earlier SAFIR projects. The motivation for writing a new tool was to make it faster and more accessible. Python was chosen as the programming lanquage of the project due to the rich open source scientific computing ecosystem available for Python. Unlike Pyroplot, which used genetic algorithms for optimizing the model parameters, PyroPython aims to be agnostic to the choice of the optimizer. Currently, the software supports Bayesian Optimization (BO), optimization by multistart method using a derivative-free optimization algorithm (Nelder-Mead), differential evolution, and random sampling. Usage of BO for parameter identification is a novel aspect of the PyroPython tool. It involves fitting a response surface model to model evalua-tions and using the response surface to explore promising solution candidates.

Several optimization methods were tested on a very challenging 16 parameter py-rolysis model fitting problem. It was found that, at least for the present optimization problem, the traditional optimization methods, Nelder-Mead simplex and differential evolution have better performance than the Bayesian Optimization methodology. This conclusion may, however, change if the optimization problem at hand would be more costly to evaluate, say a long cone calorimeter experiment or a bench scale experiment.

The participation to PRISME 3 was continued. The project will provide high quality, large-scale experimental data on the topics that are relevant to fire safety of nuclear power plants. These results can be utilized directly in the safety assessments, or for simulating and validating the simulation tools.

Deliverables in 2018

- In Task 1.1 it was demonstrated that kinetic parameters for continuum-scale pyrolysis model can be derived from atomistic simulations. The work was documented as a VTT research report.
- In task 2.1, three different heat transfer models were evaluated to predict cold-side temperatures of a fire barrier element. An Aalto research report was written.
- In task 3.2, a new tool ('PyroPython') was developed as a successor for the Pyroplot tool for pyrolysis model parameter identification. Development of the tool is described in a VTT research report.
- The PyroPython software is available online at https://github.com/PyroId/PyroPython. The PyroPython documentation is available at http://pyroid.github.io/

2.3.3 FOUND - Analysis of fatigue and other cumulative ageing to extend lifetime

Project FOUND concerns cross-disciplinary assessment of ageing mechanisms for safe management and extension of operational plant lifetime. It develops deterministic, probabilistic and risk informed approaches in computational and experimental analyses with education of new experts. It consists of seven scientific work packages (WPs): WP1 Remaining lifetime and long term operation of components having defects; WP2 Susceptibility of BWR RPV internals to degradation mechanisms, including a dissertation; WP3 Fatigue usage of primary circuit, with emphasis on environmental effects; WP4 Fatigue and crack growth caused by thermal loads; WP5 Development of RI-ISI methodologies; WP6 Dynamic loading of NPP piping systems; and WP7 Residual stress and their relaxation in NPPs.

Specific goals, results and deliverables in 2018

Each of the work packages of the project had separate and distinct goals as discussed below. The deliverables of each work package are presented directly after the chapter.

WP1 was dedicated to investigation of criticality of defects found in NPP components. With the modern developing NDE methods, more indications are found year by year. The work in 2018 considered the evaluation of allowable flaw sizes in piping components according to the ASME section XI rules. The rules are applicable in determining the acceptability of flaws found by examinations but the methods can also be applied in determining the maximum allowable flaw size for a given component under known loads. In the WP, two ASME XI procedures (those in Appendix C and Appendix H) were compared against the FFS approach given in BS7910. The results were presented as an ASME PVP conference paper (D1.1.1).

The deliverable of this work package is:

- A conference paper on the study on the comparison of ASME XI and BS7910 allowable flaw sizes for piping components.

WP2 provides an investigation on the susceptibility of BWR RPV internals and their supporting structures to various relevant degradation mechanisms. The aim of the four-year work was to prepare a dissertation on the subject. The work consists of a literature review, covering available relevant literature and databases and of a set of computational analyses, including development of new computational applications. The computational part covers both deterministic and probabilistic approaches. The scope included computational assessment of the propagation of degradation in the susceptible BWR RPV internals, providing new computational developments for assessment of the propagation of degradation and providing conclusions on degradation potential of BWR RPV internals. The dissertation manuscript is the main deliverable and also some related publications were prepared:

- Dissertation manuscript concerning the degradation potential of BWR RPV and its internals
- An ASME PVP conference paper on the long-term operation of BWR RPV
- A research report on the stress and degradation analyses of the RPV pump deck weld.

WP3 studying the fatigue usage of primary circuit aims to educate new experts and gain practical knowhow and learn of international progress and challenges related to the transferability of laboratory fatigue data to primary circuit fatigue assessment and usage monitoring. Strain-controlled fatigue experiments using the FaBello facility in hot and pressurised reactor coolant water were performed (Figure 2.38). International networking was achieved through participation and presenting in scientific forums. The main objective of WP3 is to reveal the underlying mechanisms and develop a model to quantify the effects of hot water environment in fatigue of stainless steel. Limitations of the current F_{en} methodologies were identified with test results, which suggest that the plastic strain rate is a more relevant parameter than total strain rate to characterize the environmental effects on fatigue life. A new Fen model was proposed based on the results obtained during the four years.

The deliverables of this work package are:

- A research report on the results of strain-controlled fatigue tests in PWR water environment.
- A conference publication on direct strain-controlled fatigue testing in simulated PWR water

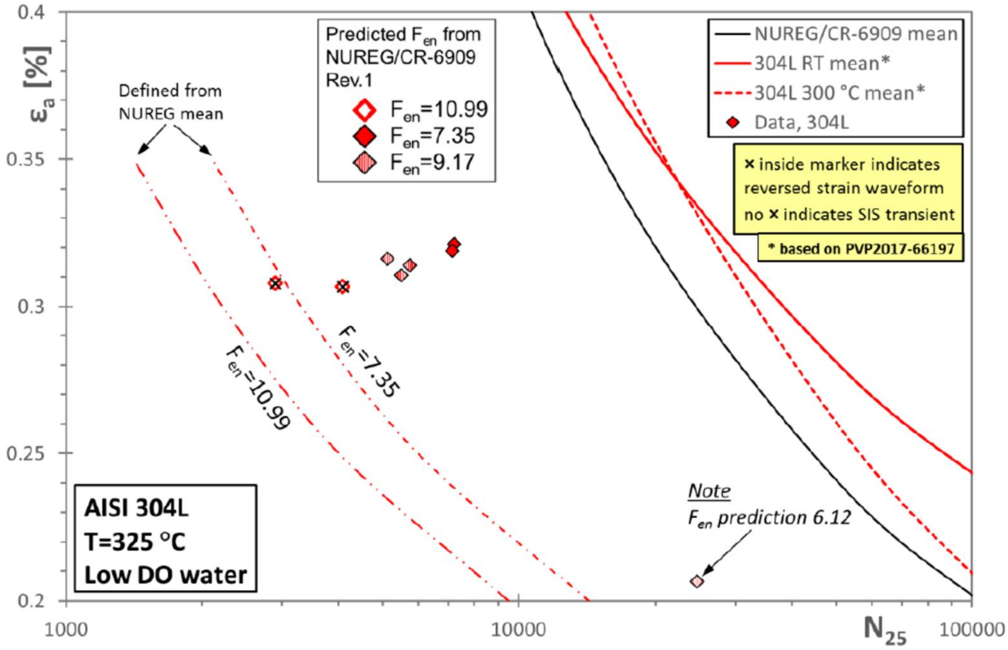


Figure 2.38. Fatigue test results with AISI 304L in PWR water compared to the NUREG/CR-6909 mean curve and nominal expected Fen.

WP4 concerns the thermal and structural evaluation of mixing loads and their effects on the degradation of piping components. Thermal mixing of hot and cold water with large temperature differences has led to fatigue and crack growth e.g. in piping T-junctions. This phenomenon is modelled using the spectrum method where an artificial temperature fluctuation signal is generated based on a given spectrum. The spectrum method was developed further by considering temperature spectra published for various T-junctions. The model spectrum was compared with measured and simulated spectra and the corresponding reference frequencies were determined. One-dimensional fatigue and crack growth calculations were performed for the Civaux T-junction, and results from using sinusoidal and spectrum thermal loads were compared. Furthermore, an ASME PVP conference paper was prepared on the subject.

In addition, developments in the modelling of a crack behaviour under cyclic thermal loads were made. A two-phase weight function method combining the surface crack and subsurface crack weight functions was shown to be able to capture crack closure behaviour possible under cyclic thermal loads. The behaviour was verified with finite element analysis with crack face contact. A journal article manuscript was prepared based on the findings.

The deliverables of this work package include:

- A conference paper on the application of the spectrum method in thermal fatigue.
- A research report on the further development on the spectrum method considering various measured and simulated temperature spectra.
- A journal article manuscript on the surface crack closure behaviour under cyclic thermal loads.

WP5 provides further development of risk assessment procedures, as well as a supplementation and update of the quantitative VTT RI-ISI analysis procedure. The VTT RI-ISI procedure is a combination of deterministic and probabilistic flaw and degradation assessment tools for the evaluation of pipe failure possibilities and the risk informed planning of inspections. The procedure allows e.g. the quantitative evaluation of the influence of different inspection intervals and inspection capability on the leak probability. The procedure combines deterministic fracture mechanical models describing the crack growth with probabilistic models to evaluate the failure probability and effect of inspections. In 2018, the VTT RI-ISI procedure was extended to include LBB assessments, where specific margins on the leak rate, crack size and load bearing capacity must be met in order to qualify as an LBB case. An example leak and break probability plot obtained with the VTT RI-ISI procedure is shown in Figure 2.39.

Another research topic addresses the connection between RI-ISI and PRA. Probabilistic risk assessment (PRA) is used to calculate the quantitative risk of nuclear accident and to analyse the importance of different systems and components. PRA's main purpose is to support risk-informed decision making. PRA also supports RI-ISI analyses by quantifying the consequences of pipe failures. There is much to be gained from better connection and mutual support between PRA and RI-ISI. One possibility to bring RI-ISI and PRA closer would be to develop a software support for the better integration. Even common analysis software is a possibility. In addition, it would be beneficial to develop an automatic piping failure consequence calculator into the PRA software. Consequently, the research introduced new RI-ISI features to calculate CCDPs and CLERPs in PRA software FinPSA. It is demonstrated how CCDP and CLERP values of piping component failures can be calculated automatically. The RI-ISI feature enables detailed modelling of the consequences of the piping component failures without complication the PRA model itself.

The deliverables of this work package are:

- A research report detailing the implementation of LBB features in VTT RI-ISI toolkit.
- A research report on the computation of consequences of piping component failures in PRA software

WP6 assesses piping systems subjected to dynamic loads. In 2018, the work considered the development of methods used to assess fast pressure induced loads in piping components. This work is concerned with a numerical implementation of fluid-structure-interaction for two dimensional piping structures. The procedure combines the method of characteristic for the fluid equations with the finite element method for beam structures. The fluid-structure-interaction is governed by the junction coupling at the pipe elbow, and the friction coupling is taken into account as well. A piping application was prepared to solve the dynamic bending stress response due to a water hammer event caused by a rapid valve closure. Comparisons of the results obtained with the fluid-structure-interaction procedure with the selected reference results are presented in the research report.

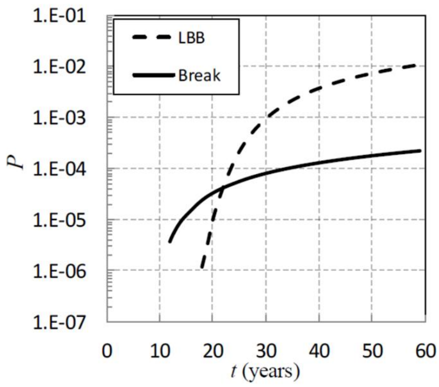


Figure 2.39. Cumulative leak and break probabilities as a function of time for an example piping elbow component.

The ASME III NB-3652 primary stress limit equation requires conservative combination of pipe moments for deterministic analyses. However, the probability of moment maximums of dynamic load cases occurring at the same time may be fairly low. Hence, the deterministic approach of applying always the most severe moment combination can be inadequate for risk informed in-service inspection analyses. The probabilistic combination of pipe moments was studied in this WP. A simplified analysis of the probabilities of the moment peaks overlap was first made. The probability density functions (PDFs) for random moment signals and for more realistic piping moments signals were then determined. The PDFs and the cumulative distribution functions (CDFs) of the resultant moments of these moment signals were determined. Resultant moment distributions from combining all time step combinations showed that most of the values are moderate. However, varying the load case initiation times and searching the maximum resultant moments produced distributions that were clustered closer to the largest possible value. For varying load case initiation times, the 50th and 90th percentile resultant moment ratios were in the range 0.33-0.81 and 0.45-0.87, respectively, for various types of moment signals. The algebraic and SRSS combination methods showed qualitatively fairly similar probability distributions. As the number of combined load cases was increased, the probability of obtaining the highest possible resultant moment decreased.

The ASME III NB-3652 primary stress limit equation also requires stress indices to take into account the effect of component geometry. The stress indices are needed for e.g. piping elbows. The ASME code provides simplified formulae for calculating the stress indices for various components based on geometric parameters. Earlier studies indicate that the code values of the B2 stress index may be overly conservative. The B2 stress index for elbows was studied. A literature survey was first conducted and less conservative B2 formulae proposed in the literature were reviewed. Geometrically non-linear elastic-plastic finite element (FE) calculations were then performed for three different elbows to determine the collapse loads and the corresponding B2 values. A mesh sensitivity study was first performed, and different ways for determining the B2 values from the FE results were then considered. The most reliable method seems to be the comparison of collapse loads of the component and of the corresponding straight pipe section. The collapse loads were determined by using the twice-elastic-slope method. The stress indexes obtained by the FE calculations were compared with the ASME code values and with less conservative correlations proposed in the literature. The FE results showed lower stress indexes compared to the code equation, and the results were in fairly good agreement with previous calculations. When using the collapse load ratio of the elbow and straight pipe, the reduction factors were from 0.58 to 0.63.

The deliverables of this work package are:

- A research report on the coupled FSI water hammer assessment of a piping elbow
- A research report on the probabilistic load combination methods
- A research report on the evaluation of B2 stress indices

WP7 develops residual stresses measurement techniques. Residual stresses play a major role in stress corrosion cracking (SCC), which is identified as significant degradation mechanism for various BWR and PWR components. During the lifetime of the existing plants, several SCC cases have been reported in the dissimilar metal welds. These have led weld repairs and replacement of susceptible materials. In particular, Alloy 82 and Alloy 182 weld materials have been replaced by the more resistant Alloy 52 and Alloy 152 materials. In the new plant designs, the weld geometry and structure has also been developed and traditional weld geometry has largely been replaced with narrow-gap welds that significantly reduce the weld material. While the change of material to less susceptible alloys have significantly reduced susceptibility to SCC, it is still of interest to study the residual stress states of these various weld geometries and material combinations. The residual stresses may still play a role in other failure mechanisms (e.g. fatigue) and affect the design of the plant. The welds are rather complex, and thus the availability of representative welds to destructive residual stress measurements is rather limited. In this report, residual stress measurements two weld samples are reported. The samples represent the traditional (with Alloy 182) and more modern narrow-gap (with Alloy 52) geometries. The narrow gap weld shows an overall good residuals stress structure; the highest tensile stresses are contained within the weld material and the sensitive root area experiences lower stresses. Tensile stresses (Figure 2.40) peak around 420 MPa, which is in-line with previous measurements on old weld geometries.

The deliverables of this work package are:

- The results and a report on the measurements of residual stresses from BWR welds removed from service
- A comparison of test samples and residual stresses from BWR welds removed from service

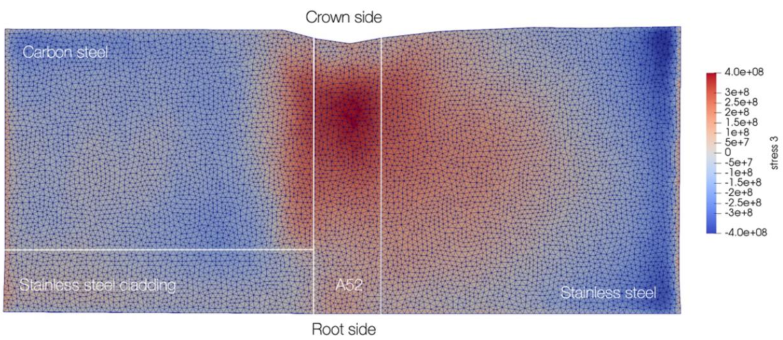


Figure 2.40. Contour results from Alloy 52 narrow-gap mock-up.

2.3.4 LOST - Long term operation aspects of structural integrity

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.

Goals and results in 2018

The YVL guide E.4 requires that the risk for fast fracture is assessed in the upper shelf region. Fast fracture in the upper shelf area can occur by brittle frac-ture or ductile fracture. In the former case, the material has not been in the actual upper shelf temperature zone. For example, rapid cooling has caused a shift in the ductile-to-brittle regime, and thus, brittle fracture can occur in the upper shelf area. In the latter case, rapid cooling can lower the ductile fracture toughness properties, which can lead to an unstable ductile fracture event. The fracture toughness in the upper shelf area is determined with J-R curves.

Before the project, there was no data, nor predictions, regarding the J-R curve development during a decreasing temperature transient. In LOST, the effect of rapid cooling on tearing resistance (J-R curves) was investigated [1]. The testing was performed in a temperature range above the range covered by the 5 % master curve, identified as the upper shelf region.

Fracture toughness specimens were cooled from 300 °C to the room temper-ature at a cooling rate of 2 °C/s. The cooling rate of 2 °C/s was determined at the center location of the specimen. The cooling rate is faster close to the surface. The cooling rate was selected to be in the same range as during a loss of cool-ant accident close to the surface. The following was concluded: For the first time, a testing method was used for characterisation of the effect of cooling rate on fracture toughness in the upper shelf region. No drastic fast fracture in the upper shelf region was observed for the investigated material during a fast cooling transient. However, the conclusion was based on the obtained load-displacement data and further analysis are required. The work continues in SAFIR2022.

The future investigations should focus on 1) the developed method should be applied for an actual RPV steel, since the temperature dependence of the inves-tigated material is not the same as for RPV steels 2) the testing method shall be developed further 3) review the cooling rate during a loss of coolant accident and do a detailed analysis of the applied cooling rate in the thickness direction of the specimen.

In addition, previous work on characterisation of decommissioned reactor pressure vessels were reviewed. Based on previous work reported in open literature, the following was concluded: In almost all of the previous investigations, the materials characterisation included chemical, hardness, macrostructure, and fracture toughness analysis in the through-thickness direction. These analyses are necessary to be able to explain the variation of material properties in thickness direction. Yet, the analyses were not comprehensive enough to be able to draw definitive conclusions from the data. A more detailed analysis of the microstructure can be applied for excluding the effect of certain phenomena, such as e.g. the damage observed in the material, and thus, improving the conclusion. The surveillance results were not reported. It is likely that the plants did not even have a surveillance programme. The Barsebäck 2 material investigated in BREDA contains surveillance samples and the material investigation methods are more extensive than in the previous projects.

In 2018, a mock-up with two girth welds and a weld inlay was studied [4]. The first girth weld was a dissimilar metal weld (DMW). The mock-up resembled a real nozzle. The target was to compare computed and measured stresses. The residual stress finite element computations were performed with axisymmetric models. The computational results showed tensile hoop stresses along the outer surface in the DMW area before inlay welding. The order of magnitude of tensile stresses is 400 MPa. The tensile stresses were reduced due to the inlay welding but the stresses remain tensile (order of magnitude approximately 200 MPa). The computed axial stresses along the outer surface were tensile after girth welding and change partly to compressive due to inlay welding.

The characterization methods were developed for DMWs. A model was developed to describe the dependence between crack location and fracture toughness. The model can be used for predicting the fracture toughness properties in the case the DMW contains a weak region and fracture progresses along this weak region. The model is based on a previous theory developed for homogeneous materials. The applicability of the previous theory was tested for a material consisting of a soft and a hard material. The theory works as long as the mechanical properties of the softer material are applied.

In 2018 a new software to predict ductile crack growth was developed. Special element that have been developed so far and coded with Fortran via user defined subroutine in Abaqus for two-dimensional models is Q16. One special sub-element is developed as well, that being element QS6. For all possible pairs of the main elements and sub-elements, the pair of Q16 and QS6 are programmed first. The number of nodes in the outer edge of Q16 is minimum, and the number of inner nodes is enough to create higher order interpolation functions for accurate prediction of the void growth and shape. On the other hand, one edge of element QS6 has a quadratic shape function, which is suitable for round edges and must be located on the solid void interface. Thus, the pair of Q16 and QS6 reduces the size of the global stiffness matrix and the higher order shape functions are placed around the void, which is the most crucial location.

Deliverables in 2018

- A journal article on a crack-location correction for T_0 analysis of an Alloy 52 dissimilar metal weld.
- A conference article on characterization of J-R curves of a HSLA-steel and an Alloy 52 DMW with SE(T) specimens
- A research report on development of special numerical approach for damage mechanical analysis with implementation in Abaqus.
- A research report on weld repair residual stress simulations and comparison to experimental measurements.
- A research report on constraint and fracture toughness
- Master’s Thesis on mechanical properties of DMWs.
- A conference article on assessment of Master Curve material inhomogeneity using small data sets.
- A research report on pre-investigation for BREDA–various aspects related to aging of the reactor pressure vessel
- Research report on applicability of HRR stress field ahead of a crack at an interface between a soft and a hard material

2.3.5 MOCCA - Mitigation of cracking through advanced water chemistry

Corrosion problems in the PWR secondary circuit are mostly related to deposition of magnetite into steam generator (SG) and enrichment of impurities into crevices within the circuit. The enrichment is typically driven by boiling. Water entering the crevices within a SG (e.g. between tube and tubesheet or under a magnetite deposit on a straight tube) boils letting volatile species escape as steam and leaving non-volatile species (salts, lead, copper etc) in the small water volume of the crevice. After some time of operation, the crevice chemistry can become very aggressive (either acidic or basic) due to impurity enrichment.
This project focuses on advanced water chemistry tools by which the formation of magnetite particulates in the feed water line can be mitigated and their deposition into SG can be minimised. To that end, the mechanism of lead assisted stress corrosion cracking as a major threat to SG integrity is researched. In addition, substitutes for using hydrazine, a potentially cancerous chemical used in PWRs, are studied.

Specific goals in 2018

The use of hydrazine, an oxygen scavenger routinely used in PWRs/WWERs both during outage and power operation to ensure low oxygen concentration and thereby low corrosion rates, is under consideration because it's negative effects on environment and health. In 2018, the target was to compare experimentally the effectiveness of several alternatives to hydrazine as oxygen scavengers. Especially alternatives which are active at low temperature, at T = 50°C, i.e. during plant shut-down were of interest here.

One of the primary causes of SG corrosion damage is magnetite particle formation in the secondary side feed water line and further deposition of magnetite particles into SGs – thus, finding ways to mitigate deposition of magnetite into the SG is a major goal in this study. In 2018, the goal was to verify experimentally the surface charge of stainless steel (a major structural material used in SG internals) in PWR secondary side water as a function of temperature.

Another clearly established cause of SG corrosion damage is the lead assisted stress corrosion cracking, PbSCC. In 2018 the target was to finalise the work on PbSCC of carbon steel in alkaline crevice environment, representative of some steam generators.

International co-operation was continued in 2018 by e.g. attending the 17th Nordic Corrosion Congress held in Copenhagen, Denmark, the Nuclear Plant Chemistry (NPC 2018) conference held in San Francisco, USA and by taking part in the European Co-operative Group on Corrosion Monitoring (ECG-COMON) work and the yearly meeting held in Ljubljana, Slovenia.

Alternatives for hydrazine

The primary requirements for an oxygen scavenger are strong reducing action and good passivation ability. Strong reducing action keeps oxygen levels entering the steam generator low, thus minimising the risk of SG tube degradation by localised corrosion modes. Good passivation ability, on the other hand, reduces propensity to flow assisted corrosion (FAC) in the feed water line and thereby reduces rate of delivery of corrosion particles into SG where they could form deposits and enhance susceptibility to localised corrosion modes.

Based on a previous literature study, the present investigation focused on the effectiveness of several hydrazine alternatives as oxygen scavengers. The conditions used were representative of a steam generator under revision period, i.e. T = 50°C and alkaline pH. To evaluate the oxygen removal rate, measurements of dissolved oxygen concentration with time in borate buffer solutions were performed and analysed with different concentrations of carbohydrazide, diethyl-hydroxylamine, methyl-ethyl-ketoxime and iso-ascorbic acid.

A comparison of the oxygen removal rate by i-AA and MEKO is shown as an example in Figure 2.41. Based on quantitative evaluation of the data, the following ranking of the studied compounds by oxygen scavenging ability at 50 °C and pH 9.2 can be proposed: hydrazine > iso-ascorbic acid > carbohydrazide >> DEHA > MEKO, Table 2.5. This is the same ranking found earlier at 21°C. The rate of oxygen removal by hydrazine was about twice higher than that of i-AA and CBH, and about seven and fifteen times higher than that of DEHA and MEKO, respectively.

The effect of the investigated additives on the corrosion of 22K carbon steel in borate buffer solution at 50 °C was also studied. Based on the data it can be concluded that all the additives increase the general corrosion rate of 22K steel (by stabilising Fe_3O_4 over FeOOH and Fe_2O_3) with respect to that in pure buffer solution, hydrazine having the strongest effect.

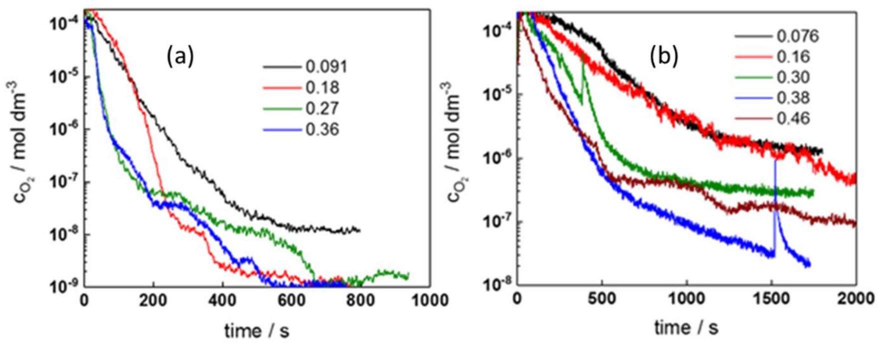


Figure 2.41 Evolution of the dissolved oxygen concentration with time in a borate buffer solution with the addition of different concentrations (in mmol dm^{-3}) of (a) i-AA and (b) DEHA at 50 °C. The rate of decrease of oxygen is clearly faster with i-AA than with DEHA.

Table 2.5. Oxygen removal rate at the highest initial scavenger concentration in mmol dm^{-3} (in parentheses).

Scavenger	Oxygen removal rate at T = 50°C / mmol dm^{-3} s^{-1}
Carbohydrazide	0.020 (0.034)
DEHA	0.0054 (0.46)
MEKO	0.0028 (0.75)
Iso-ascorbic acid	0.025 (0.45)
Hydrazine	0.043 (0.32)

Magnetite and stainless steel surface charge

During previous years in the current project we have studied the effect of ammonia, morpholine, ethanolamine and octadecylamine (ODA) on the surface charge of magnetite particles. This year, the target was to study the effect of temperature on the surface charge of stainless steel. However, due to equipment malfunction and failure, this target was not reached.

Mechanism of lead assisted stress corrosion cracking of carbon steel

Lead (Pb) has been detected in effectively all tube-support samples, crevice deposits and surface scales removed from steam generators. Typical concentrations seen are 100 to 500 ppm but in some plants, concentrations as high as 2,000 to 10,000 ppm have been detected. The cracking susceptibility is believed to have a strong dependence on the redox-potential of the crevice environment. Redox-potential, on the other hand, is affected by the amount of e.g. copper oxide in the crevice solution.

The PWR steam generator tube materials considered to be most resistive towards stress corrosion cracking (SCC), i.e. Alloy 600TT, Alloy 800 and Alloy 690 have each been shown to be susceptible to SCC enhanced by the presence of lead (PbSCC). In case of VVER-type PWRs, the steam generator tubing is most often stainless steel which has a relatively low susceptibility to PbSCC. However, the VVER steam generator body material, carbon steel, has been very little studied, in spite of wall-through cracking insidents found both in Russia and Czech Republic.

The technique for studying SCC in this project was to perform slow strain rate tests (SSRT) in which a tensile specimen is loaded at a constant strain rate until fracture occurs. Susceptibility to SCC is deduced from the reduction in fracture strain as compared to that in an inert environment (e.g. air at the same temperature) and additionally from the morphology of the fracture surface.

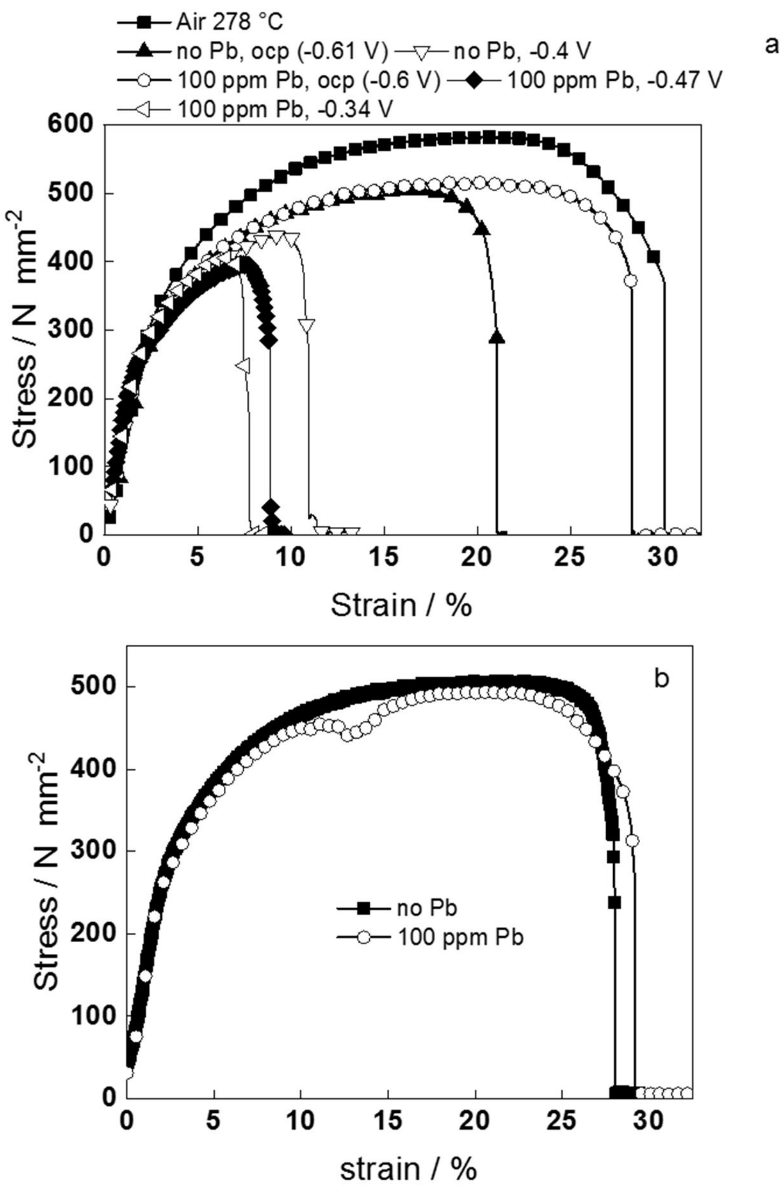


Figure 2.42. Slow strain rate testing (SSRT) results: (a) at open circuit and at several anodic potentials in acidic crevice solution, (b) at open circuit in alkaline crevice solution with and without Pb addition.

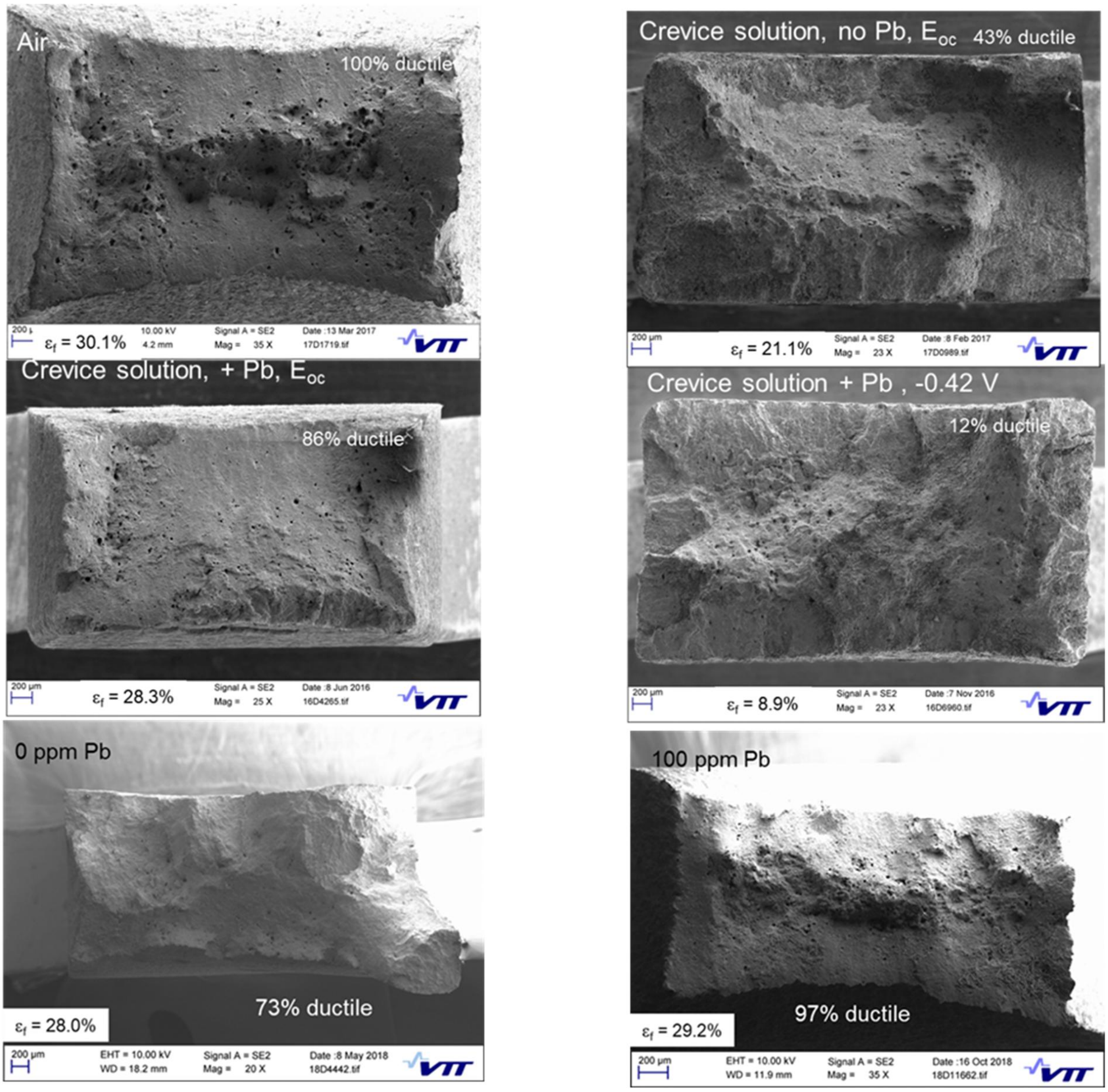


Figure 2.43. Micrographs of cracked SSRT specimens in air (above, left), acidic crevice solution without (above, right) and with Pb at open circuit (middle, left) and at -0.42 V(middle, right), (below) corresponding micrographs in alkaline crevice solution at open circuit without (left) and with Pb (right).

Figure 2.42a shows a comparison of stress-strain curves of of carbon steel 22K in crevice solution at T = 278^{o}C with and without Pb, at corrosion potential and at slightly elevated potentials. Surprisingly, addition of 100 ppm Pb at corrosion potential is seen to increase the fracture strain from 22% to 28%, i.e. make the material more ductile. However, increasing potential (simulating slightly oxidising environment produced by e.g. oxygen inleakage into the SG) results in a dramatic reduction of the fracture strain to values at and below 10%. Figure 2.42b shows similar curves measured in alkaline crevice solution. Without Pb, the fracture strain is 28.0% and with Pb 29.2%, indicating a rather small susceptibility to SCC under the alkaline conditions. The fracture surfaces shown in Figure 2.43 corroborate the information found in the SSRT in that the percentage of ductile fracture surface (i.e. area with no marks of SCC) is clearly higher for alkaline solution than for acidic solution, with or without Pb.

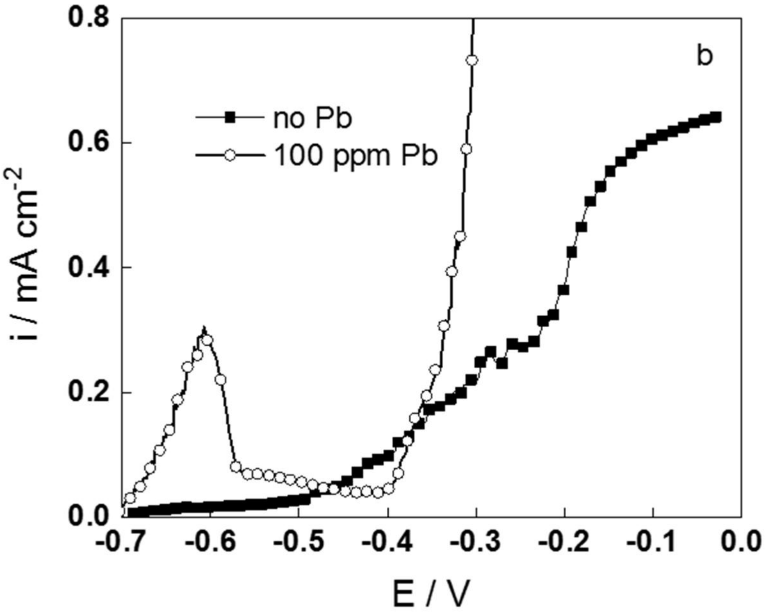


Figure 2.44. Current density vs. potential curve for carbon steel 22K in simulated alkaline SG crevice environment at 278 °C with and without the addition of 0.45 mmol kg^{-1}(100 ppm) Pb as PbO.

The current density - voltage -curve in the alkaline crevice environment without lead showed a slowly increasing trend as a function of potential, indicating a passive surface, Figure 2.44. The rate of increase goes slightly up above -0.35 V_{SHE}, at which potential Fe_3O_4 (magnetite) is expected to transform to Fe_2O_3 (heamatite). The curve in the presence of Pb, on the other hand, shows an active peak around -0.62 V_{SHE}, followed by passivation and another increase in current density for potentials higher than -0.4 V_{SHE}. Based on these results it can be assumed that close to the corrosion potential (-0.7 V_{SHE}) Pb^{2+} ions are reduced on the steel to metallic Pb. At potentials more positive than the corrosion potential, Pb starts to dissolve as $PbCl^+$ accounting for the active peak in the current density - potential -curve in Figure 2.44. The dissolution of lead is most probably accompanied with Fe dissolution leading to an unstable passivation of the surface. The reduction in the SCC susceptibility in presence of Pb (as noted by higher fracture strain in Figure 2.43 and higher portion of ductile area in Figure 2.44) is proposed to result from this unstable passivation, which at lest partially prevents localisation of corrosion necessary for SCC to occur.

2.3.6 THELMA - Thermal ageing and EAC research for plant life management

The objective of THELMA is to perform investigations that increase the understanding of factors affecting thermal ageing and environmentally assisted cracking (EAC). Development of techniques and utilisation of versatile techniques to investigate the phenomena is central in THELMA. Through characterisation of different degradation phenomena in nuclear materials from safety relevant components, knowledge is gained on how the materials behave during long-term operation in LWRs. This is essential knowledge to be used in failure analysis, safety assessment, time limiting ageing analysis, maintenance, lifetime management, material selection and maximising the lessons learned from plant events.

Specific goals in 2018

Work package 1 (WP1) focus on thermal ageing of austenitic nuclear materials, i.e., weld metals, cast stainless steel and Alloy 690. The investigations in 2018 has focussed on cast stainless steels and publication of the most recent results on Alloy 690. Double-loop electropotentiokinetic reactivation DL-EPR measurements on thermally plant-aged cast stainless steel (SS) from hot and cold leg of Ringhals steam generator were performed with different scan rates and solution strength. The DL-EPR results show an increasing sensitivity of the measurement with slower scan rate, Figure 2.45. However, the results are not straightforward and comparable to literature results on materials aged at higher temperature. The cold leg material, with lower thermal load, and lower hardness show a higher I_r/I_a ratio than the hot leg material, which is opposite to literature observations. This shows the complexity of thermal ageing, comprising of several mechanisms, which are dependent of time and temperature. It also shows that although DL-EPR seems to be applicable to materials aged at higher temperatures, it is not directly applicable to plant aged materials.

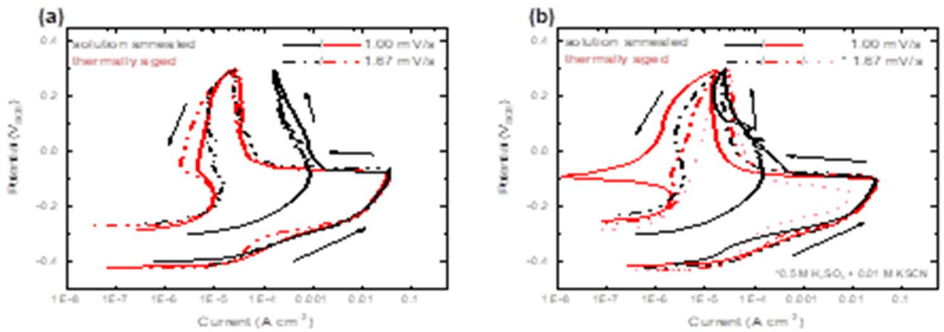


Figure 2.45 The electrochemical response is stronger with slower scan rate in both the (a) hot leg and (b) cold leg materials. The response vs thermal ageing is not straight forward, as it is for laboratory aged materials.

A literature study on VVER high Ni-materials was performed as basis for planning future investigations. The available literature is scarce and completely lacks information on the metallurgical state of high-Ni VVER materials. That would be essential information when analysing the long-term behaviour of the components, i.e. possible changes in intermetallic phases, possible irradiation effects etc. Further basic metallurgical characterization of XH35BT and XH77ЮР alloys were recommended, in order to clarify the metallurgical state and microstructure typical of plant components for both materials. Such investigations will be started in a small project ordered by the SAFIR2018 management board in 2019.

To increase the understanding of factors affecting irradiation assisted stress corrosion cracking (IASCC), characterisation of irradiated SS materials were performed as part of the OECD Halden project. The materials, i.e. Type 304L SS with a dose of 5.9 dpa, cold worked Type 316 SS (9 dpa) and cold worked 316Ti (4 dpa) show similarities, but also differences in the deformation structure. The amount of clear channels was highest in Type 304L, while the localisation of deformation was most severe in CW Type 316. Evidence of epsilon martensite was detected in the 304L material, being the least stable material, but not in the others. The precipitates in the Ti-stabilised material were surrounded by deformation bands, which may affect cracking. The amount of deformation defects was smallest in this materials, indicating that the precipitates act as sinks for the defects. The crack growth rate of the 5.9 dpa 304L was similar to that of the BWROG disposition line for NWC, while that of the CW316 material was one order of magnitude higher. These results shall be taken into account when selecting materials for e.g., bolts and assessing the long-term behaviour of existing components subjected to irradiation.

WP 2 deals with understanding initiation and precursors for environmentally assisted cracking and determination of corrosion fatigue assessment, through participating in, and reporting the development of two EU-projects, i.e., EU-INCEFA+ (Increasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment) and EU-MEACTOS (Mitigation of EAC Through Optimisation of Surfaces). So far, in total 94 fatigue tests have been completed in INCEFA+ on the common material (304L) and 15 on national materials. The tests results show a clear effect of environment on fatigue life. The effect of surface roughness is more pronounced in air than in PWR environment. The effect of hold time during strain control is not apparent at low strain amplitude, while literature data show an effect at higher strain amplitude, with unloading during the hold. The progress in MEACTOS has been preparation of a state-of-the-art report, preparation of a test matrix and preparation of the test materials. The test materials are Alloy 182 and SS 316L, focussing on long-term operation and new builds, respectively. The surface conditions are industrial (milling), grinding and peening. Testing will commence in 2019, starting with screening tests and continuing with evaluation tests.

WP3 focus on irradiation embrittlement of reactor pressure vessel (RPV) materials, and is done in co-operation with the SAFIR2018 LOST project and the Swedish BREDA project, dealing with trepans cut from the Barsebäck 2 RPV. Both the THELMA and LOST parts will be incorporated into the BRUTE project in SAFIR2022. The objective is to determine how the microstructure affect the mechanical properties and onset of brittle fracture. In 2018, microstructural investigations on non-irradiated B2 materials after mechanical testing were continued from 2017. The primary initiation site were determined, and cross-sections close to the initiation site investigated. The results show that initiation at large secondary particles lower the fracture toughness (FT), as does initiation in the reheated weld metal compared to initiation in as-welded weld metal. Pre-fatigue fracture surface topography affects also the FT value. The FEG-TEM investigations on B2 weld metal show three types of secondary particles, with an average size of 257 nm.

International co-operation and knowledge transfer include participation in international networks and presentation of the THELMA results at international conferences. The PM is member of the steering committee of the ICG-EAC group and member of the international scientific board for the Fontevraud conference. The ICFG-G_EAC meeting in 2020 will be arranged in Finland, and preparatory work is ongoing. International meetings are important for knowledge transfer to Finland, and are excellent networking and learning arenas for the young generation.

Deliverables in 2018

Scientific publication were written on the results from the investigations on thermally aged cast stainless steel, and thermally aged Alloy 690. Thermal ageing results in short range ordering, increase in hardness and in lattice contraction. A good correlation is shown between nanoindentation, lattice contraction and microstructure.

The results from the ATEM investigations on neutron irradiated SS were presented at the Fontevraud 9 conference in France. The presentation received good visibility, and increased our knowledge, but also our position as a major laboratory for post irradiation examinations in CNS.

Thermal ageing affect the uptake of hydrogen in Alloy 690, as shown in performed measurements using thermal desorption and mechanical loss spectroscopy. Results were presented at the SteelyHydrogen conference in Belgium.

A key note lecture summarising the results from the thesis by Roman Mouginot on thermal ageing of Alloy 690 was given by prof. H. Hänninen at the Eurocorr conference including suggestions for further investigations. The role of hydrogen on primary water stress corrosion cracking in combination with thermal ageing and short range ordering is one of the topics suggested for further studies.

Yearly progress reports of the EU projects INCEFA+ and MEACOS were written for knowledge transfer to SAFIR2018 RG5 on these topics.

A detailed research report summarise all microstructural investigations performed on the B2 RPV materials. The results will be used when similar investigations are done on irradiated materials later. The goal to tighten the coupling between mechanical behaviour and microstructure has been met.

Optimisation of methods to investigate high boron SS was studied and reported, showing that Electron Energy Loss Spectroscopy (EELS), ATEM and wave length spectroscopy (SEM/WDS) all give additional information about the complex microstructure and the presence of boron, and that specimen preparation of materials is challenging. Similar investigation are planned on irradiated materials, and the gained insights in specimen preparation and technique sensitivities will be used in the coming work.

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

Passive components in nuclear power plants (NPP)s are monitored through in-service inspections (ISI)s. These passive components consists for example primary circuit 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.

Specific goals in 2018

Main focus for WP1 was to extend virtual flaw capabilities through simulated flaws. Semi-elliptical flaws with surface roughness was simulated using CIVA software and implemented in virtual flaw software, where the flaw signal was altered the similar way as for real thermal fatigue flaws. Unfortunately, the simulated flaws seemed to stand out from the thermal fatigue flaws when a POD measurement was conducted, thus indicating that the simulation was not accurate for human inspectors to use. While the human inspectors could not tell whether the flaw they found was simulated or not it gives hope for these simulated flaws. Thus next step is to generate flaw signals more accurately, for example using finite element models.

Other approach in WP1 was the quantitative NDE. Preliminary tests were done in order to determine Poisson's ratio through ultrasonic measurements. The test showed promise since the ratio was able to be measured with non-contact method with the same accuracy as a conventional contact method. This allows options to study ultrasonic behaviour in materials such as welds and also opportunity to study material degradation non-destructively.

WP1 has participated in PIONIC project and the participation will continue in SAFIR2022 in RACOON project.

In WP2 the main objective of the whole project has been a constructing the mock-up wall for research and teaching purposes. The construction of the mock-up wall was finished Fall 2018.

A final report of the construction of the wall was written, consisting info on the methods, flaws and measurement systems used and implemented to the wall. The construction opens whole new research opportunities for concrete NDE research years to come.

In addition to the wall construction WP2 has participated in international ODOBA project and hosted a special guest lecture regarding Non-destructive evaluation for Nuclear Power Plant concrete infrastructure held by Rsearch Professor Dr. Niederleithinger from BAM.

Deliverables in 2018

- Short research report of using V(z) mothod to characterize austenitic stainless steel weld.
- A conference paper submitted to 12th ECNDT 2018 about a Hit/Miss POD with simulated and emulated flaws.
- A research report of the construction of the concrete wall mock-up.
- A research report of the evaluation and calibration of NDT&E methods.

Deliverables from year 2017

- A research report of the condition zero of the mock up wall and detection of embedded defects.
- A research report of the current state-of-the-art in using probability of detection in the field of reinforced concrete structures condition assessment.
- A research report of the condition zero of the NDE and monitoring methods for the concrete mock-up wall.

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

Different polymer based materials are widely used in various applications in nuclear power plants and inside containments, e.g. cable jacketing/insulators, sealants, paint coatings, lubricants and greases. As any other material or component, polymers are susceptible to ageing. Elevated temperature, ionizing radiation and moisture are considered to be the most important ageing stressors and they tend to interact with the polymer structure in different ways. In addition to these ageing stressors, the properties of polymer blend, e.g. degree of crystallinity, amount of fillers and antioxidants, has an effect to the ageing behaviour. Thus, the degradation mechanism can be quite complex.

COMRADE was developed on input from a feasibility studies from Energiforsk AB and STUK and through discussions between VTT, RISE and the Nordic NPPs through Energiforsk. When developing COMRADE, it was understood that there are gaps in knowledge for setting functional based acceptance criteria at the nuclear power plants. Furthermore, a need in gaining a better understanding on how a polymeric component reacts to different levels of low dose ionizing radiation and synergistic effects between thermo-oxidative and radiation degradation was identified.

Specific goals in 2018

The work in COMRADE was divided in three different work packages (WPs) where each WP has their own specific goals. In WP1 the goal was to terminate accelerated ageing test on FKM and Nitrile O-rings and from the test data obtained, to suggest preliminary acceptance criterion for this specific O-ring design. Also the FEM model development, where the model could predict acceptance criterion as function of the O-ring cord size, was to be terminated. In WP2 the goal was to acquire polymer components from the operating Nordic NPPs and study their properties. In WP3 various polymer ageing related issues was studied. First, related to the modelling work conducted with polymers, the goal was to continue the development work aiming to model and understand the reverse temperature effect on XLPE. Secondly, the experimental work concentrated on studying dose rate effects on EPDM and Lipalon cable jacketing material by applying semi-empirical superposition model. Thirdly, the aim was to further develop the ToF SIMS technique to measure oxidation profiles induced by ageing on sample surfaces.

During 2018 ageing test series were performed on FKM (Fluorine rubber, often known under the trade name Viton) and Nitrile rubber. Test sheet and O-rings were exposed and analysed according to the same test schedule used for the EPDM tests except for the exposure temperatures used in oven ageing. The FKM (Viton) O-rings showed no leakage after exposure and the mechanical properties were not significantly affected. Figure 2.46 below show compression set for the FKM material after exposure to different ageing temperatures. Compression Set values of nearly 100% did not correspond to leakage.

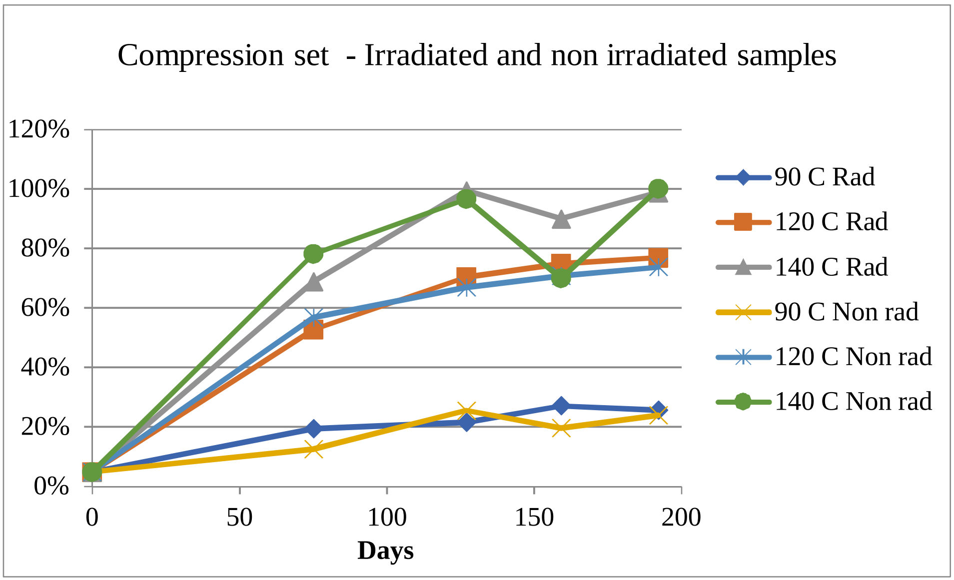


Figure 2.46: Compression Set values for FKM rubber.

Nitrile rubber showed somewhat more degradation compared to FKM. At elevated exposure temperatures all mechanical properties increased significantly initially. Thermogravimetric Analysis showed that the material contained low molecular additives which may evaporate at elevated temperatures. Figure 2.47 below shows compression Set as a function of exposure time at different temperatures, Compression set at start was measured by compressing a standardized test specimen at 23°C for one week. Another explanation for initial deterioration of mechanical properties is could be additional crosslinking at elevated temperatures.

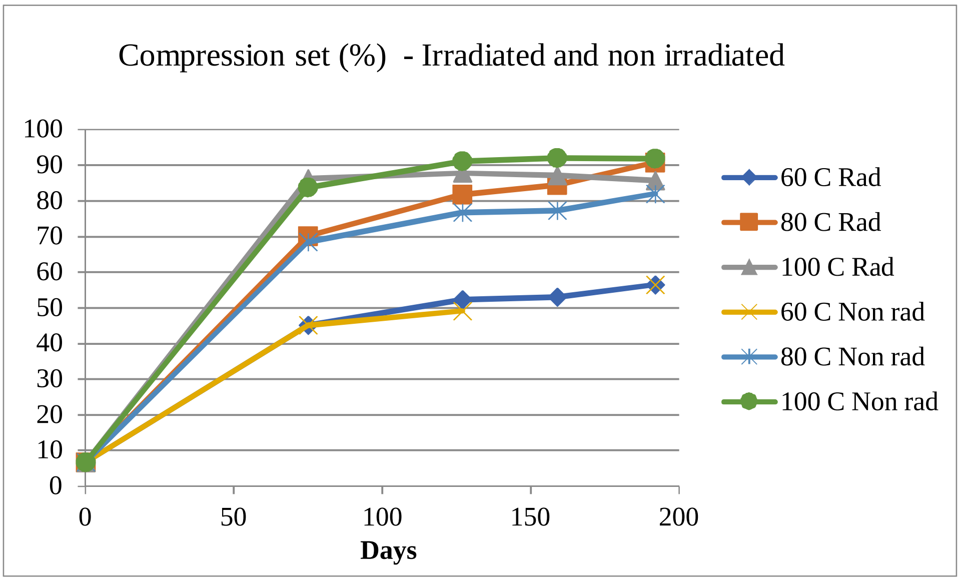


Figure 2.47: After an initial increase of compression set the values remain more or less the same during ageing.

The highest compression set values were around 90% at 100°C and no leakage was observed. Hardness had increased significantly from around 80 IRHD to 99 IRHD which is on the high end of the IRHD scale and almost not possible to measure. Like for the EPDM rubber, FKM and Nitrile were not affected by the moderate doses of gamma rediation used in this project.

Usually end of life of a sealing is usually said to be at a compression set at 50%. Leakage tests show that the set is much higher before leakage actually occurs. However the effect of other disturbances as for examples vibrations is not included in this study and must be taken into account when setting end of life criteria. Also, the O-ring function at elevated temperatures may be necessary to study to set a final acceptance criteria. Hardness is a simple and convenient measuring technique to use on site but the reason for increased hardness must be well investigated, i.e. is hardness increase caused by loss of plasticizer, crosslinking or oxidation. Different degradation affects the sealing performance differently.

For the FEM modelling on material properties Stress relaxation was also measured on EPDM and from these results a model was calculated. The model was calibrated using compression set data and the final model corresponded well to the measured data.

In WP2, the mapping of available polymer materials were continued and the focus was on the currently operating plants. A questionnaire was drew up which was sent to power plant contact persons. The questionnaire compiled information on available polymer components, their service and storage history. A number of O-ring samples were analysed at RISE during 2018 but since the number of each sample was very limited hardness was the only method possible to use was hardness. Many of them were very degraded and were also damaged when they were dismantled. Hardness measured varied between 70 IRHD, probably close to initial value and between 90-99 IRHD indicating degradation. According to the sealing measurements in WP1 these O-rings may still function. To be able to analyse exposed material planning of the outtakes is essential in terms of type of sample, amount of sample and available documentation.

WP3 content can be divided into three different subtasks where the first one focuses on using computational modelling techniques in polymer ageing. Based on the results obtained during 2016 literature survey modelling work is focused on explaining the “reverse temperature effect”. In 2017 proper modelling framework was developed with molecular dynamics (MD) simulations to conduct this. In 2018, the modelling work of WP3 was started by performing an extensive set of simulations on the crystallisation of cross-linked PE. The main finding was that after crystallisation, cross-links were found exclusively in the amorphous regions. A method was developed and tested for simulated aging of XLPE structures, which involves both bond scission and cross-link formation. Simulations of the tensile test were carried out for PE systems with varying chain length and varying cross-link density. A method to include stretch-induced bond breaking was developed for the united-atom description. According to results, the mechanical strength of polyethylene decreased with decreasing chain length until the material became fragile. Conversely, an increasing cross-link density first improved the mechanical strength, but at high cross-link densities the material became increasingly brittle, as evidenced by a decrease in the elongation at break. These results were in qualitative agreement with the experimentally observed reverse temperature effect, as originally reported by Celina et al (Radiat. Phys. Chem. 48 (1996) 613).

The second subtask is related to developing technique measuring oxidation profiles on aged EPDM samples. During 2016 three different techniques were compared in oxidation profile measurements. ToF-SIMS seemed to be the most promising technique and the measurement procedure was further developed in 2017. The surface and bulk oxidation of EPDM were studied as function of absorbed dose and dose rate. ToF-SIMS seemed to be very sensitive technique in detecting oxidation products in the samples but identifying artefacts (caused by e.g. uneven surface and contamination during handling) from the normalized signals still causes uncertainty to the measurement results.

The third subtask focused on studying EPDM and CSM in thermal and irradiation environments. From the obtained test results, the following conclusions were made. Firstly, based on the thermal ageing data and calculated activation energy value, estimated remaining lifetimes for CSM samples at 50°C and 25°C can be provided (ca. 680 days and ca. 16 years, respectively). Secondly, the sequence of thermal and irradiation treatments in ageing of the studied EPDM blend matter (see Figure 2.48). Thirdly, the degradation levels in EPDM samples were too low after the ageing treatments that dose rate effect analysis could be conducted. Fourthly, sufficient degradation levels were obtained with CSM samples, but still the dose rate effect analysis conducted based on superposition would not yielded in reliable results due to the narrow test matrix. These observations show that the predictive semi-empirical models based on superposition require a lot of experimental data to function properly.

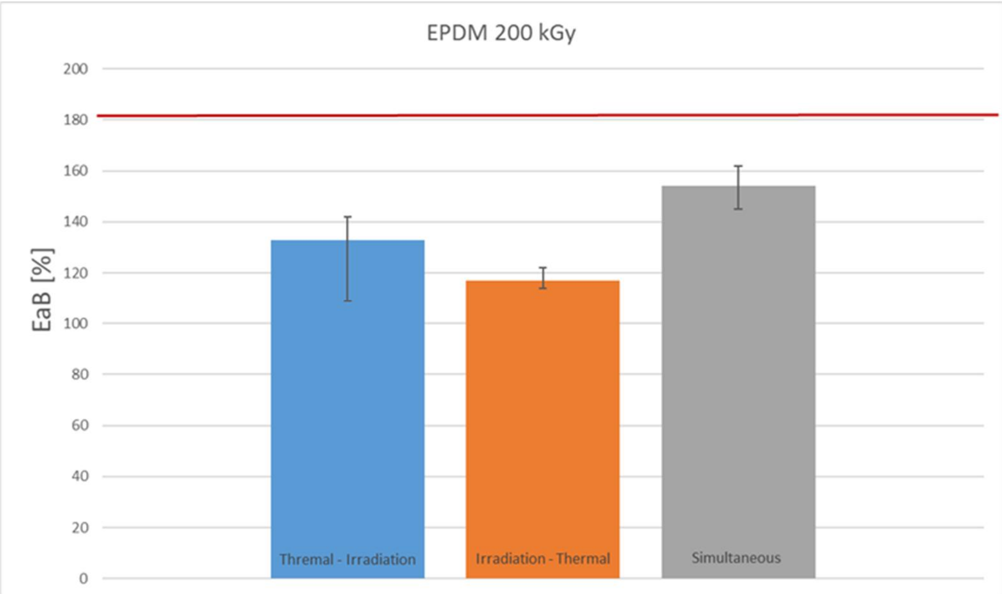


Figure 2.48: Comparison of sequence of ageing on EPDM samples. Blue column describes the decrease in EaB when sequence is thermal-irradiation, orange column irradiation-thermal and grey column simultaneous thermal and irradiation ageing.

WP4 focused on international cooperation. In 2018 COMRADE workshop was organized in Forsmark, Sweden by Forsmark NPP. The two-day meeting was held at the Forsmark NPP during 4th and 5th of December. The first day concluded from lectures given by various instances from NPPs, regulator, polymer industries with topics relevant to ageing of polymer components in nuclear power plants. The second day focused presenting the project results, discussions and presenting the future research project proposal. In addition, visits to the local final repository site were organized.

Deliverables in 2018

- Development of condition monitoring technique for O-rings used in NPP applications (presentation at Fontevraud 9 conference)
- Development of FEM model and test run using data from T1.1 (RISE-research report)
- Conduct workshops on each powerplant to discuss results and how to use them (minutes of the meeting)
- Material acquisition from plants and description how to include the materials in project work (RISE-research report)
- Scientific publication on the molecular dynamics modelling efforts to understand the reverse temperature effect (peer reviewed publication)
- Development of ToF SIMS technique in determination of oxidation gradient on EDPM (conference publication in Fontevraud 9)
- Application of semi-empirical ageing techniques on EPDM and Lipalon cable jacketing materials (VTT-Research report)
- COMRADE Workshop/seminar on polymer ageing issues at NPPs (minutes of the meeting)

2.3.9 EVOGY - Evolving the Fennoscandian GMPEs

The broader aim of the SAFIR/EVOGY and the embedded SYNTAGMA/NKS projects is to maintain a network of experts focused on diffuse seismicity areas of the Nordic Countries and further enhance the cooperation between seismology and earthquake engineering. Such cross-disciplinary cooperation supports the power utilities and regulators by providing background information for the safety assessments of nuclear power plants (NPP).

The EVOGY project was targeted with a clear outcome and duration of only one year. Hence, recruiting and educating new experts was not feasible in this timeframe, besides the Institute of Seismology beginning the training of a new expert by launching a new BSc thesis on the topic.

The objective of the project is to evolve the Fennoscandian Ground Motion Prediction Equations (GMPEs) to a state-of-the-art format, giving the utilities and the regulator an effective tool for assessing the seismic hazard of the NPPs. The project included (i) analysis of the newly acquired datasets from recent earthquakes and synthetic ground motions generated in the SYNTAGMA project, (ii) implementation of the most modern intensity measures in the analysis procedure, and (iii) calibration of the GMPE constants.

Specific goals in 2018

In Task1 the goal was to create a homogeneous dataset of earthquake recordings for the calibration of the GMPE. We carried out quality control of the recorded data by making old and newly added data more uniform in terms of their format. ISUH and VTT worked together within this task, creating synergy and internal review to the process. We created seven consecutive updates of the dataset, gradually improving the procedures of data processing. The dataset contains recordings of Fennoscandian earthquakes with magnitude above 2. Task1 provided a uniformly processed earthquake dataset of the measurements for Fennoscandian earthquakes up to 2017, data to be released as Deliverable 1 of EVOGY.

The goal of Task2 was to implement a calculation procedure for spectra calculation and link it to the earthquake data. ISUH and VTT have agreed on the data interfaces, and the spectral calculation procedure is automated. Hence, additional ground motion data can easily be accommodated and the spectra re-calculated. An independent checking protocol was implemented to test the steps of the automated routines, to assure the correctness of the spectra calculations.

The automated methodology developed in Task2 was used to obtain the database of the spectra for all the collected ground motions, practically all Fennoscandian earthquakes with magnitude above 2. The spectra database will be released in excel format by the project as Deliverable 2 and 3.

Finally, we used these spectra to calibrate the constants of the proposed GMPE. The calibration constants and measures of the prediction's error margins are reported in the Final report of the EVOGY project (Deliverable 4a) and a manuscript in preparation (Deliverable 4b). Some examples of fit between the GMPE prediction and selected Fennoscandian earthquake recordings are given in Figure 2.49.

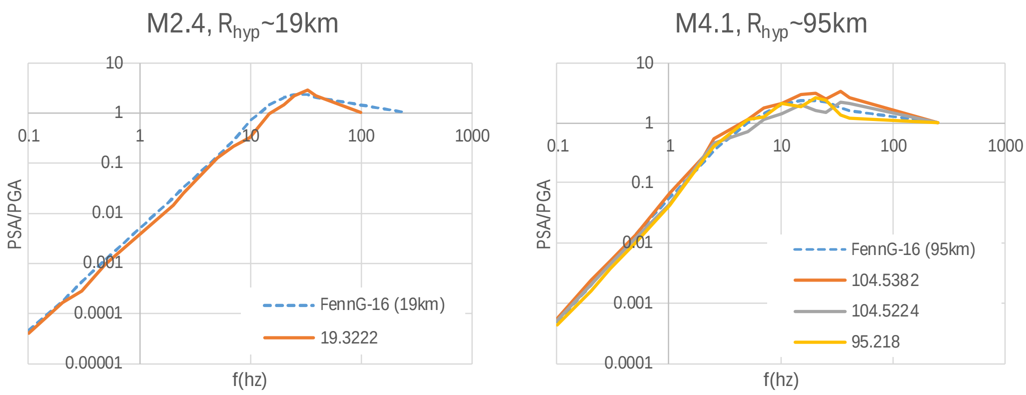


Figure 2.49. Comparison with selected spectra from the Fennoscandian dataset a) from a 9/27/2008 event with depth=17.3km, b) from the Gulf of Bothnia event of 2016-03-19 with depth=23.5km

As an embedded activity to the SAFIR/EVOGY, in the NKS/SYTAGMA project we used our hybrid modelling method[1] to generate larger datasets of synthetic ground motions. The fit of

[1] Fülöp, Ludovic, Vilho Jussila, Björn Lund, Billy Fälth, Peter Voss, Jari Puttonen, and Jouni Saari. 2017. Modelling as a Tool to Augment Ground Motion Data in Regions of Diffuse Seismicity - Final Report. NKS Nordic Nuclear Safety Research.

the synthetic data to the near-field of the GMPE prediction is shown in Figure 2.50, while the displacement field of the earthquake is shown in Figure 2.51.

At this point we formulated the GMPE only using natural earthquake recordings, we only accepted the synthetic recordings to indicatively guide the short distance tale of the GMPE. In this range, there are no natural recordings, so it is debatable how the spectra evolve very close to the fault. However, we plan to check the developed GMPE with merged natural-synthetic data sets.

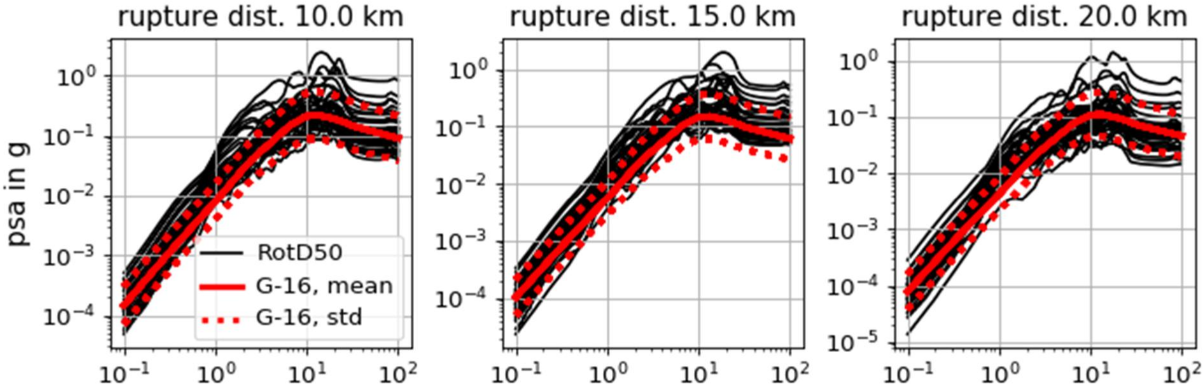


Figure 2.50. Comparison of spectra in black from the simulated synthetic ground-motion and the GMPE in red. The plot corresponds to M_w=5.0.

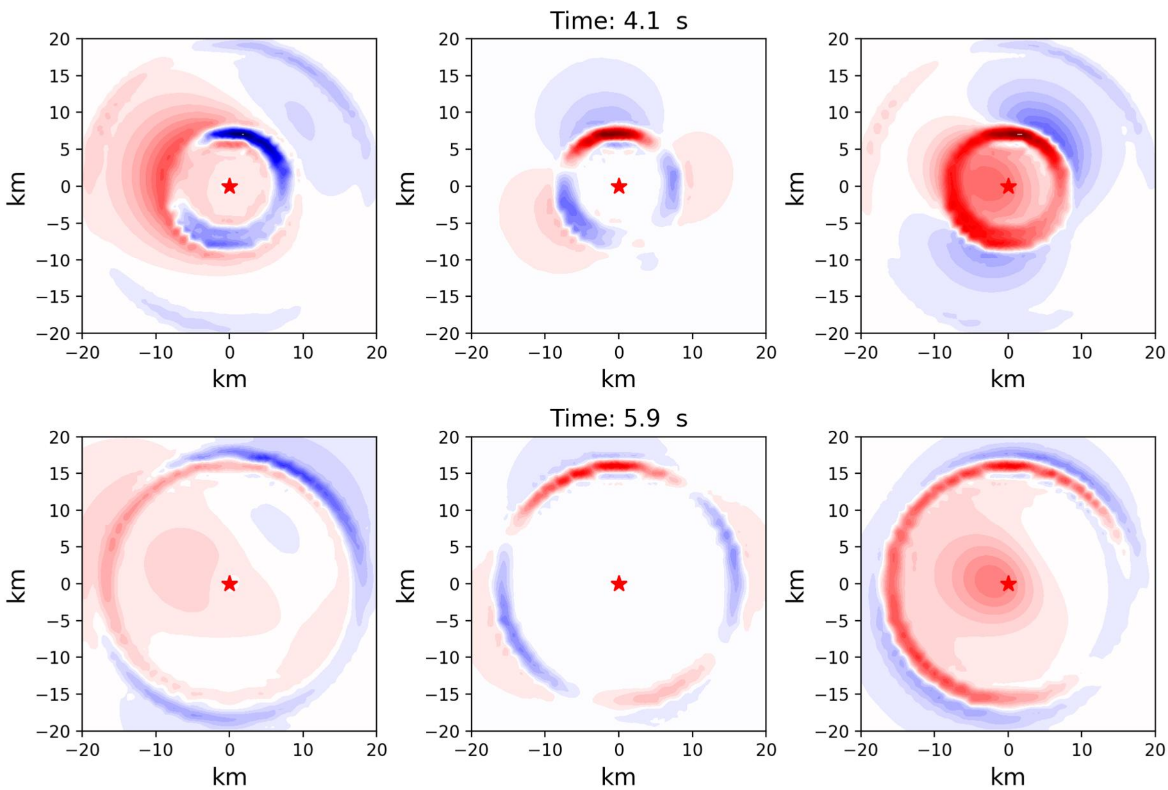


Figure 2.51. Displacements (mm) on the ground surface from an M_w=5.0 earthquake, occurring on a fault dipping 45 degrees, in the case of oblique faulting. The rows in the figure are time slices at 4.1 and 5.9 s. The columns in the figure are the radial from epicentre, transverse and vertical components of the displacement. The depth of the hypocentre is 10 km.

Deliverables in 2018

- Integrated data-sets of earthquake recordings. Ready – will be delivered to power companies and STUK on an EVOGY HDD/DVD.
- Calculation of spectral and new intensity measures. The calculated spectra is in EXCEL worksheet and will be delivered on an EVOGY HDD/DVD
- The code was used to calculate the spectra and it is written in Python.
- Research paper draft, with a shortened version to become final Report for the SAFIR book was delivered 08.02.2019 for review to Fennovoima (Juho Helander)
- NKS report draft ready. Still in review with authors. Agreed to be sent to STUK (Pekka Välikangas / Simon Burck).

2.4 Research infrastructure

In 2018 the research area “Research infrastructure” consisted of four projects:

1. Development of thermal-hydraulic infrastructure at LUT (INFRAL)
2. JHR collaboration & Melodie follow-up (JHR)
3. Radiological laboratory commissioning (RADLAB)
4. Barsebäck RPV material used for true evaluation of embrittlement (BRUTE).

2.4.1 INFRAL - Development of thermal-hydraulic infrastructure at LUT

The general objective of the INFRAL project was to develop the thermal hydraulic measurement infrastructure of the LUT University nuclear safety research laboratory. The up-to-date experimental research infrastructure is essential for the modern nuclear safety analyses. The implementation of novel measurement techniques in the thermal hydraulic experiments is needed for the validation of the Computational (Multi-)Fluid Dynamics (C(M)FD) methods. Important part of the INFRAL project was the further development of the techniques related to the advanced measurements and their applications. The goal was to build good in-house expertise in the use of recently acquired techniques to facilitate the needs of computational modellers in the future experiments in the best way if it is technically possible. The CFD grade measurements can give new insights into the physics behind the different flow phenomena that may ultimately lead in the improvements in the safety of nuclear power plants. Furthermore, the goal of the project was to secure the operability of (PWR) PACTEL and other test facilities, as well as to launch a study on the new major test facility to prepare for the post-PACTEL era.

Specific goals in 2018

In 2018, the INFRAL project was divided into four different work packages. The first work package (Advanced measurement techniques) included activities that are related to the use of advanced measurement techniques at LUT. Part of the work was to develop analytical tools to extract the needed data from the measurements. The other part was to study the applicability of the techniques for different flow problems and to develop new measurement solutions. The second work package (Maintenance and equipment) aimed on the maintenance of (PWR) PACTEL and other test facilities, and it comprised the yearly inspections, calibrations etc. The third work package (Modular Integral Test Facility (MOTEL)) aimed on designing and constructing a new large-scale integral test facility in the LUT laboratory. The fourth work package (Project management, international co-operation and publications) included the tasks related to the project management and participation to the reference group meetings and seminars. Also, international co-operation actions, such as research visits, were a part of the work package.

The work package 1 of INFRAL consisted of research topics that are related to the study and application of the so-called advanced measurement techniques: Particle Image Velocimetry (PIV), Wire-Mesh Sensors (WMSs) and High-Speed Cameras (HSCs). The measurement systems were acquired to LUT already during the previous project (ELAINE) in 2011–2014. During the SAFIR2018 research programme, the advanced measurement systems were developed further and used in multiple applications. In 2018, various activities were carried out to strengthen the in-house expertise and the know-how related to the measurement systems. It is essential that the researchers are familiar with the equipment and can also acknowledge the possible limitations. Some of the application targets of the measurement systems are also related to non-SAFIR research projects.

The PIV measurement system was utilized actively in 2018 for contract measurements exclusively. The follow-up to water flow in a rectangular channel was executed during the first half of 2018. A new method of seeding the flow with natural convection flow next to a heated wall was successfully designed and executed. The PIV system in action is presented in Figure 2.52. In addition, PIV was used in wind tunnel measurements. These measurements focused on the drag reduction by small groove riblets on surface of a wing. The measurements were conducted together with the Laboratory of Fluid Dynamics at LUT University. New experience considering seeding of gaseous flows was gathered. In addition, data from Constant Temperature Anemometry (CTA) was gathered for velocity comparison, and a publication from the results is in the making. In 2018, there were no hardware or software updates for the PIV system.

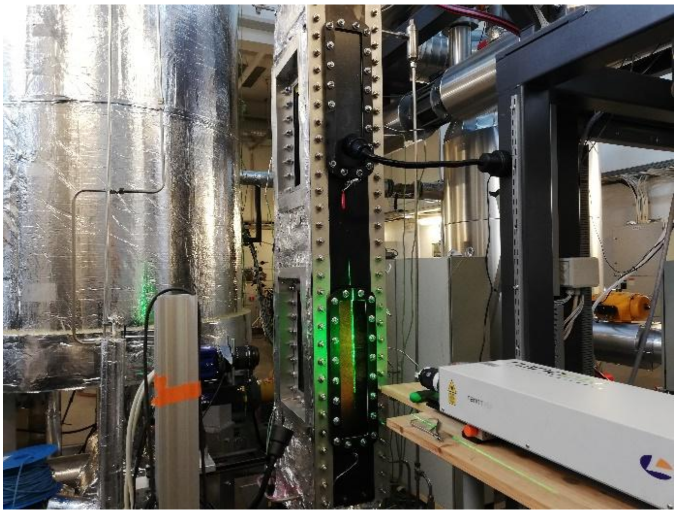


Figure 2.52. *The PIV measurement system in action in planar-PIV mode with new sCMOS cameras.*

PIV has been found out to be an effective tool in velocity measurements where it is applicable. PIV has been applied multiple times with success in many projects, but studies within SAFIR programme have been challenging stated with the reasons mentioned before. However, a lot of important user experience has been gathered within SAFIR-related studies. All in all, the usability of the PIV system has improved substantially during the INFRAL project. PIV will remain an important part of the LUT thermal hydraulic measurement infrastructure in the future.

The advanced applications of the wire-mesh sensor technique have been actively studied at LUT. The axial sensor design (AXE) was designed and constructed in the previous ELAINE project in the SAFIR2014 programme to tackle the problems related to the measurement of the axial flow behavior. During the SAFIR2018 programme, the functioning of the sensor was studied.

In 2017, the axial WMS technique was studied further with measurements in swirling two-phase flow. In 2018, the swirling two-phase flow WMS measurement results were further analysed, and a conference paper was prepared concerning the results. An abstract titled "*Defining Void Fraction with Axial Wire-Mesh Sensor in a Swirling Two-Phase Flow*" was sent to SWINTH-2019 (Specialists Workshop on Advanced Instrumentation and Measurement Techniques for Experiments Related to Nuclear Reactor Thermal Hydraulics and Severe Accidents, 22–25 October 2019, Livorno, Italy). The abstract was accepted in September. The full paper was prepared during late 2018, and it will be finalized during the first half of 2019. The main focus of the paper is to evaluate the performance of the axial sensor under swirling flow conditions, particularly the AXE sensor's ability to determine void fraction distributions compared to traditional radial sensors.

The conclusion of the analysis of the axial WMS measurement results is that because of the intrusive structure of the axial sensor design, it cannot measure void fraction distributions very accurately. Especially in the case of swirling flow, the upstream edge of the sensor breaks the flow and affects the void fraction results. Thus, initial design for a new kind of sensor has started in the LUT laboratory. Changing the structure of the axial sensor in a way that the intrusive edge would be eliminated completely and adding the distance between the sensor wires either with insulation material or by using different thickness circuit boards have been considered.

The high-speed cameras are used at LUT to support the data analysis of the PPOOLEX condensation experiments conducted within the INSTAB project. A pattern recognition algorithm has been developed for analyzation of the HSC measurement data. In 2018, the NURETH-17 conference paper "*Frequency Analysis of Chugging Condensation in Pressure Suppression Pool System with Pattern Recognition*" was invited to the special issue of Nuclear Engineering and Design and it was published in December 2018. In 2018, SEF-POOL tests continued. Two tests were conducted for the development of the pattern recognition algorithm. In these tests, a steam jet was horizontally directed through a sparger into a water pool, where the steam condensed by a direct contact condensation (DCC). The SEF-INF1 experiment was operated within the condensation oscillations (CO) mode and the SEF-INF2 was operated within the bubbling condensation oscillation (BCO) mode.

The SEF-INF2 was chosen as the first reference case for the pattern recognition algorithm development and CFD simulations. Snapshots of the SEF-INF2 tests 1-6 are presented in Figure 2.53. The SEF-INF2-6 experiment was chosen as the most interesting case due to its larger bubbles and BCO mode.

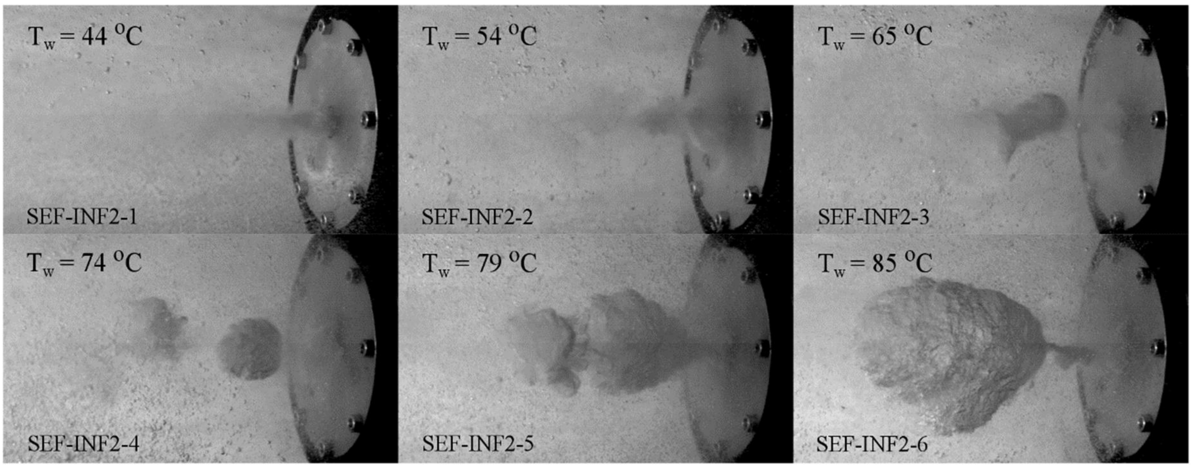


Figure 2.53. SEF-INF2-1…SEF-INF2-6 experiments. Steam mass flux was hold constant (180 kg/(m²s)), meanwhile the temperature of the condensation pool water T_w increased from the initial temperature of 12 °C to the SEF-INF2-6 tests 85 °C. It can be seen that SEF-INF2-1…SEF-INF2-3 tests are in condensation oscillation mode (T_w = 44 °C… 65 °C), SEF-INF2-4 and SEF-INF2-5 tests are near bubbling condensation oscillation mode (T_w =74 °C and 79 °C respectively) and SEF-INF2-6 in pure bubbling condensation oscillation mode (T_w = 85 °C).

Results of the image and data analysis and CFD simulations of the SEF-INF2 test will be presented in the NURETH-18 conference at Portland, USA in August 2019. An abstract titled *"Analysis of Bubbling Mode Condensation Oscillation in Horizontal Sparger"* was submitted and accepted, and the full paper will be written in early 2019.

In addition to PIV, WMSs and HSCs, development of other advanced measurement techniques was followed within the INFRAL project, too. One example of recently developed measurement techniques is the Distributed Temperature Sensor (DTS) based on Rayleigh-backscatter phenomenon. The sensor, which utilizes optic fibers, enables the measurement of temperature distribution in high detail in different geometries, such as a slab or a rod. The electronic hardware for the DTS was purchased to LUT in 2017. The functioning of the technique was planned to be conducted in 2018. The testing was originally planned to be conducted in the new passive heat removal test facility, PASI, within the INTEGRA project as one of the characterization experiments of PASI. However, the testing was dropped out from INTEGRA. The problem was, also, that the firmware of the electronics was not yet capable to handle temperature measurement. Thus, the testing of the technique was postponed to 2019.

Within the work package 2, the yearly calibrations of the (PWR) PACTEL measurements were carried out during the summer. In 2018, the heating element of the PACTEL pressurizer was also renewed because one of the heaters was broken. As the heating element was replaced, the pressurizer heating power was increased to 21 kW from the previous power of 13 kW. The upgrade of power transformers has been planned to enable higher heating power to be available for the thermal hydraulic experiments (1 MW à appr. 2.8 MW). The upgrade process has been postponed, and the options for the upgrade are still being studied within the organizations involved. In addition to the three affected laboratories of LUT School of Energy Systems, also Suomen Yliopistokiinteistöt as the owner of the buildings, Lappeenranta Energia as the power provider, as well as electrical system design consultants are part of the process.

In the work package 3, as the design of the MOTEL test facility was finalized, a report describing the first configuration of the facility was written. The first MOTEL version represents NuScale’s SMR design, which has a unique helical coil steam generator. The behavior of this type of steam generator will be one of the main interests regarding the experiments with the first MOTEL configuration. MOTEL will be assembled in the LUT laboratory during 2019. The facility will evolve in the future, and it can be upgraded with e.g. horizontal steam generators and/or an SMR containment. The first MOTEL configuration is presented in Figure 2.54. The actual design and the construction of MOTEL are funded by the Academy of Finland.

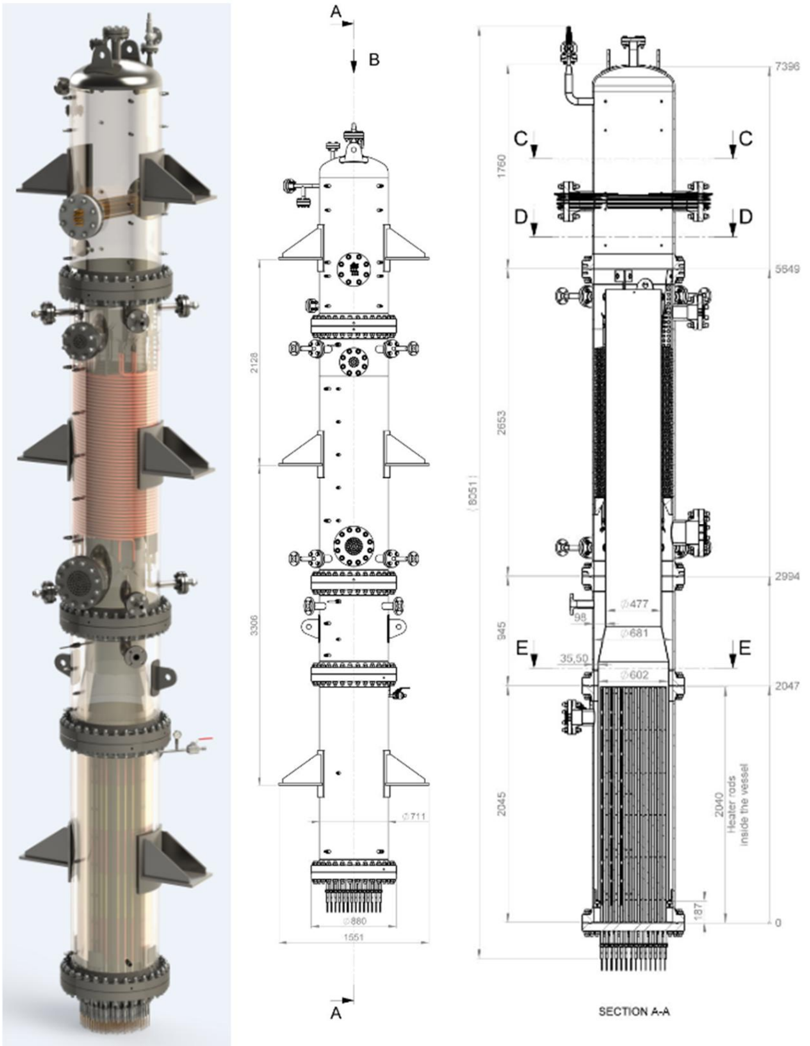


Figure 2.54. An overall view and the dimensions of the first configuration of the MOTEL test facility. The three changeable modules can be detected: core module, steam generator and pressurizer.

Within the work package 4, two researchers from LUT attended the Short Courses on Multiphase Flows at ETH Zurich, Switzerland. Profound syllabus of multiphase flows was gained. The occasion also served as a good opportunity to network with other experts on the field.

Deliverables in 2018

- A report describing the actions and development of the advanced measurement techniques (PIV, WMS, high-speed cameras etc.) in the LUT laboratory in 2018.
- A short report on maintenance, which describes the main maintenance actions in the LUT laboratory in 2018.
- A report, which describes the preliminary design of the first configuration of the forthcoming integral test facility, MOTEL.

2.4.2 JHR - JHR collaboration & Melodie follow-up

The Jules Horowitz Reactor (JHR), a new European material testing reactor (MTR), is currently under construction at CEA Cadarache research centre in France. Finland is participating in the construction with a 2 % in-kind contribution, which includes Underwater Gamma spectrometry and X-ray radiography (UGXR) and Hot-cell Gamma spectrome-try and X-ray radiography (HGXR) systems, as well as a Mechanical Loading Device for Irradiation Experiments (MeLoDIE). With this in-kind contribution, Finland will have the opportunity to use and directly benifit the new JHR research infrastructure dedicated to nuclear safety related research. Furthermore, the in-kind contribution enables access to the results of the future experiments.

The JHR consortium has set up three working groups (WG) – (i) Fuel WG, (ii) Materials WG and (iii) Technology WG - to determine experimental needs and plan future experiments. To have our national interests brought forward and to be able to follow and participate in the planning of the experiments, VTT has named participants to each of the three WGs. The WGs hold meetings twice a year, and in spring an annual JHR Technical Seminar is held, where the outcomes of the WG meetings and the progress of in-kind work are presented. The first work package (WP) of this project focuses on this collaboration through WG participation. The latter WP concerned the in-kind contribution Melodie, which was finished in 2016.

Deliverables in 2018

Travel reports:

- WG meeting 9.-10.10.2018
- Technical seminar 21.-22.3.2018
- OECD/NEA 2nd Workshop 4.-5.10.2018

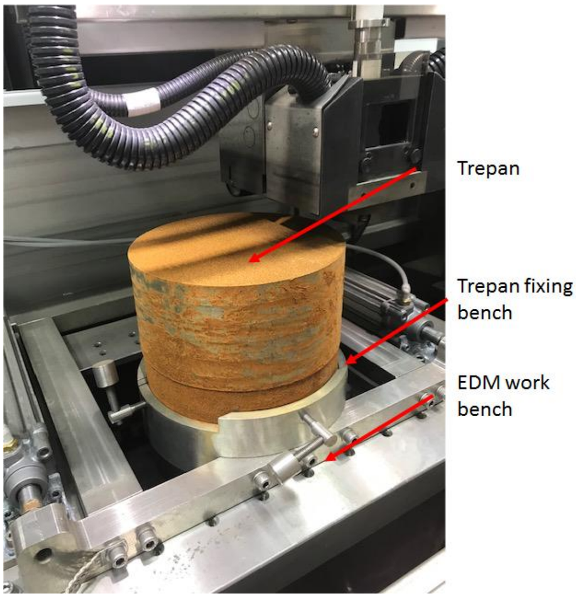
2.4.3 RADLAB - Radiological laboratory commissioning

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, the project oversaw the design and fabrication of the new hot cells, while in the second half of the program the focus was on their 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.

Specific activities in 2018

WP1 – Hot-cells: Following the manufacturing and installation of the hot cells, taking them into full use requires significant effort in developing the functionality of all in-cell equipment and the procedures around their use. In WP1 the remote operation methods and tools were developed to utilize the facilities and equipment correctly, purposefully and effectively.

Following success with getting the **in-cell EDM** working in tandem with the water cleaning circuit located in the basement, efforts continued in 2018 towards understanding and accommodating what proved to be a more complicated flow rate variability than originally expected in the different operational modes (working tank filling, emptying and water flow during cutting). Complete elimination of contamination from passing into the EDM circuitry with the set-up seems difficult to achieve though. Nonetheless, with the saw operational, trials were carried out for cutting specimens from the trepan samples for the BRUTE project.



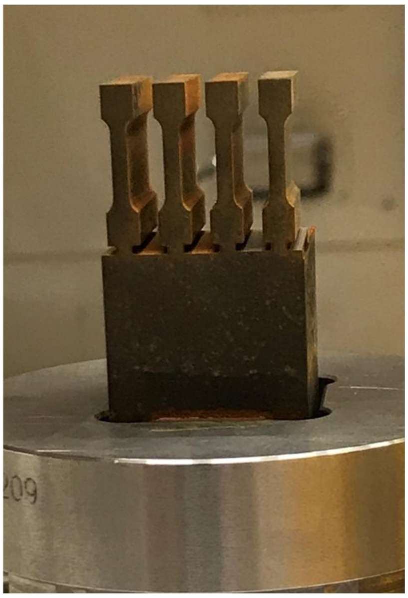


Figure 2.55. With the electric discharge machine operational, trials were carried out for cutting specimens from the trepan samples for the BRUTE project.

Regarding the **in-cell EBW**, all nuclearization components for the EBW were installed, and the supplier carried out a final day of training for VTT personnel. The device is essentially ready for normal operation.

All the **in-cell mechanical testing devices** benefited from the addition of two new employees, one an experienced testing engineer and another a research scientist focusing on fracture mechanics. As a part of preparing for device accreditation, a KMF 100 Calibrator was purchased to check the linearity of extensometers. The principle mechanical testing methods were evaluated for accreditation in end of 2018.

A few hot cell workers attended the international HOTLAB 2018 meeting in Helsinki, which featured a full session on in-cell testing methods of structural materials, and several relevant posters as well.

WP2 – Equipment procurement: This work package is mainly tasked with continuing to execute the procurement process for the hot laboratory equipment. This includes in particular investments included in the RADINFRA investment-aid instrument. A special mini-expo held at the international HOTLAB 2018 meeting featured 14 different companies, many of them suppliers of equipment and devices for hot cell environments.

In the 2018 project year road transport casks have been researched, discussions held with suppliers, and budgetary and delivery time information received. Unfortunately, it appears that the cost would be close to 3 million euros for a suitable cask, which seems unrealistic.

All of the analytical radiochemistry devices have been ordered and delivered, and trainings held. The light optical microscope (LOM) was ordered and delivered at a cost of half what was originally budgeted. The savings in the LOM are to be put towards a conventional spark OES device for measuring trace elements and carbon content in metals.

WP3 –Research equipment: This work package is mainly tasked with the development and construction of those research devices that are not readily available on the market, but rather require custom design. During 2018 the **autoclave** installations were completed and trial runs carried out. The devices for hot specimen handling progressed, but final assembly will not be completed until February 2019. The final detailed design for the custom **in-cell sawing and milling** machine proposed by Metecno was approved, and its fabrication is underway, still on schedule for installation in March 2019. The new **NPP filter testing** facility construction got underway, and the task leader estimates 50% completion this year for the total assembly; about ¾ of the needed investment procurements have been made in 2018.

WP4 – Supporting facilities: The VTT CNS requires a number of supporting facilities for handling and storage of specimens and dry and wet waste. The infra for them are realized mainly as subcontracts, and thus follow-up is mainly as a procurement process. However, implementation requires effort in areas such as producing the waste handling guidelines and population of the specimen database.

In 2018 the remaining key components of the **wet waste handling system** were fabricated by Platom Oy and installed at VTT. Simultaneously the wet-waste handling guides for the radiochemistry labs were written, and the general waste handling guide was updated for submission with paperwork for A-class laboratory licensing.

The ITD subcontract for the **specimen storage device** progressed such that the design review took place in May, the Factory Acceptance Test took place in August, and installation commenced in late September. Due to some fabrication issues, some silo tubes had to be repaired, delaying final acceptance, but all main issues were resolve by the end of December 2018; the final payment was made in January 2019. The Pergament specimen database was populated with the laboratory rooms in the facility, as well as with all of the alpha-numerical locations in the specimen storage device.

Figure 2.56. The fabrication and installation of the alphanumerically organized shielded storage device for radioactive specimens was executed in the 2018 project year.

WP5 – VTT CNS: This task is focused on taking the VTT Centre for Nuclear Safety radiological laboratory in general into use. This includes preparing the licensing papers and developing some specific laboratory methods.

Already in the first quarter of 2018 the **license** expansion papers were submitted to STUK, and the review process concluded with an understanding that there is not further obstacle to granting the license expansion, but a need for improved waste handling methods and prevention of legacy waste accumulation was identified, particularly with future decommissioning in mind.

With regard to the inductionally coupled plasma mass spectrometer (**ICP-MS)**, the round robin was completed, two posters were presented at a conference, and a third person participated in the Thermo training program.

Deliverables in 2018

- A written report describing the functionality of the in-cell Electric Discharge Machine, showing that, while working in tandem with the water cleaning circuit the device is ready for cutting things like mechanical test specimens from RPV trepan samples, complete elimination of contamination from passing into the EDM circuitry seems difficult to achieve with the current set-up.
- A written report describing the functionality of the in-cell Electron Beam Welder, showing that it is capable of joining materials for applications such as test-specimen reconstitution, although the ideal welding parameters for each type of specimen still need evaluation.
- A written report describing the functionality of the Zwick RKP 450 impact test devices with semi-automatic tempering and specimen feed, describing that it has been calibrated by VTT External Services, and is in principle ready for conducting impact testing of all materials in a locally shielded position in the laboratory.
- A written report describing the functionality of the Zwick in-cell Z250 electromechanical test device with environment chamber, demonstrating that the device functions in accordance with requirements, and therefore is ready for use in mechanical testing of various kinds, such as four-point bend, tensile and compact tension specimens.
- A written report describing the functionality of the in-cell pre-fatigue devices, including the Piezomatic device employed only for three point bend SEN(B) type specimens, and the MTS that is primarily for compact tension C(T) type specimens but also suitable for SEN(B) specimens.
- A written report describing the functionality of the OGP Smartscope CNC200 “Flash” in-cell table-top digital optical dimensioning microscope, which has been calibrated, and is ready for use in research and testing work.
- A written report describing the functionality of the Struers DuraScan-80 in-cell hardness testing device, which has been calibrated, compared with other devices via an external and internal testing round-robin, and is therefore ready for use for its intended purposes.
- A procurement specification for a certified radioactive materials road transport cask.
- Identification of three potential suppliers for the road transport cask, and cost and delivery time estimates from each of them for a cask meeting the specifications.
- Delivery of an Ortec alpha-spectrometer device procured via a legal tendering process for analyzing the specific alpha-emissions from particular radionuclides.
- Delivery of an Agilent 5110 SVDV inductionally coupled plasma optical emission spectrometer (ICP-OES) procured via a legal tendering process.
- A written report describing the functionality of the installed ICP-OES device, demonstrating that the device functions in accordance with requirements, and is therefore ready for use in assessing the isotopic composition of materials.

- Delivery of an AeroTrak 9110 aerosol particle counting device procured via a legal tendering process, for cleanroom monitoring, process tool monitoring, and filter test applications.

- Delivery of an Ultrawave digester device via a legal tendering process, for dissolving substances into a liquid prior to injection into one of the radiochemistry analysers.

- Delivery of a Zeiss Axio Observer 7 inverted metallograph light optical microscope following a legal tendering process, for installation into the hot cell for use in examining polished cross-sections of radioactive materials.

- A written report describing the functionality of the new Cormet hot autoclave facility, following a test-run using O-ring specimens, demonstrating that the new autoclave and its hot water loop are ready for conducting SCC tests in simulated PWR water conditions.

- Selection of Meteco as supplier following legal public tender for the custom in-cell sawing and milling device for installation in Cell 3.1, followed by acceptance of the engineering design for subsequent fabrication of the device.

- Drawings and fabrication of key components of the new NPP iodine filter testing system.

- Delivery and installation of the wet-waste handling liquid evaporator-recondenser designed and fabricated on a contract with Platom Oy.

- Written report describing the functionality of the wet waste handling device operation.

- Internal written guide for the Centre for Nuclear Safety laboratory waste handling approach.

- Detailed design of the physical radioactive specimen storage facility produced on a subcontract with Isotope Technologies Dresden GmbH.

- Factory acceptance test and subsequent installation and site acceptance test of the physical radioactive specimen storage facility for orderly temporary storage of research and testing materials.

- Population of the Pergament specimen database system with all the alphanumerical storage locations of the new physical storage facility, for maintaining orderly records of the radioactive specimens stored on site.

- A research report describing the results of the round robin tests of the high resolution inductionally coupled plasma mass spectrometer (HR-ICP-MS) located in the new clean room setting in the Centre for Nuclear Safety, demonstrating its efficacy over a range of elements of different substances.

- Proficiency certificate for a third operator of the HR-ICP-MS following formal training at the Thermo course in Germany.

- A conference poster on quality assurance measurements with reference materials at the 9th Nordic Conference on Plasma Spectrochemistry.

- A conference poster on elemental characterization analysis of decommissioning materials from the FiR1 Triga Mark II type research reactor, at the 9th Nordic Conference on Plasma Spectrochemistry.

- Approval from STUK for expansion of the license to operation of the full radiological facility, including the new hot cells.

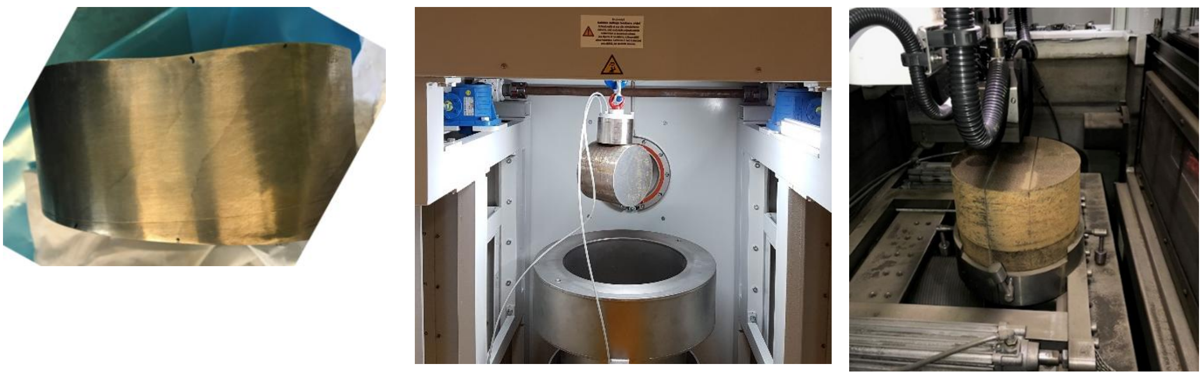
2.4.4 BRUTE - Barsebäck RPV material used for true evaluation of embrittlement

The objective of BRUTE is twofold, i.e., to prepare for mechanical and microstructural investigations of Barsebäck 2 BWR reactor pressure vessel (RPV) materials in irradiated and thermally aged condition and to pioneer the new infrastructure in the Centre for Nuclear Safety, CNS, VTT. The first objective is connected to investigations, which will be performed in SAFIR2022 BRUTE project. Eight trepans with a diameter of 200 mm and full RPV wall thickness (~130 – 160 mm) have been cut from the B2 RPV, both from the beltline welds, subjected to both thermal load and neutron irradiation during operation, as well as from the vessel head, subjected only to thermal load. Materials from surveillance programmes and from accelerated irradiation to mimic about 200 years of operation are also available. The main objective is to determine the comparability of the surveillance programs, used to assess the safety and lifetime of the RPVs with the results of the real RPV material. Pioneering the new CNS hot cell infrastructure will fully commission the new infrastructure, prove the functionality of the equipment in normal project work, confirm the quality of the results in routine testing situations and give a valuable reference for future assignments.

SAFIR2018 BRUTE project was a one year project, starting in 2018, and will continue into SAFIR2022. The work in BRUTE2018 is divided into four work packages, namely WP1 dealing with mechanical testing, WP2 with microstructural investigations, WP3 with stakeholder and dissemination issues and WP4 with transport of irradiated materials.

Specific goals in 2018

The work in WP1, mechanical properties includes development and validation of specimen preparation and test techniques. The work started with writing a report describing the work needed and the status of the infrastructure. Preparations for handling the 40 kg trepans, Figure 2.57a, exceeding the size used in the design for CNS has included lifting and movement trials using magnet lifters and dummy trepans (shaft pieces with same dimensions as the trepans), Figure 2.57b.



(a) (b) (c)

Figure 2.57. Appearance of a Æ200 mm, 40 kg trepan (a), training moving a dummy trepan in and from a transportation cask (b), and cutting of a dummy trepan using the designed jig and the new Electric Discharge Cutting machine (c).

A jig for the electric discharge machine (EDM) has been designed and manufactured, delivered, assembled and tested, Figure 2.57c. A dummy trepan has been cut into three pieces using the Electric Discharge Machine, EDM, to evaluate the cutting speed and cutting and surface quality.

Mechanical test evaluation for tensile testing and impact testing are ongoing well, aiming for accreditation of the methods at CNS. The work will continue into SAFIR2022 BRUTE.

A preliminary test plan comprising of more than 1500 test specimens has been formulated and agreed on for future optimisation and prioritisation.

The work in WP2, microstructural investigations, includes validation of such microstructural procedures, which have not been validated before, and preparation for microstructural characterization work in BRUTE, Figure 2.58. A report describing the procedures needed for the work and the status of the needed infrastructure has been published. The report describes all procedures, from cutting slices from the trepan to transmission electron microscopy (TEM)-investigations, the needed infra as well as the status of the infra. The investigations will include microstructural characterisation of the weld metal, fractography is mechanical test specimen fracture surfaces, hardness measurements from macro- to micro scale, chemical analyses, etc., Figure 2.59. Procurement of some equipment is ongoing and is a part of RADLAB and will continue in LABWAST and VTT projects.

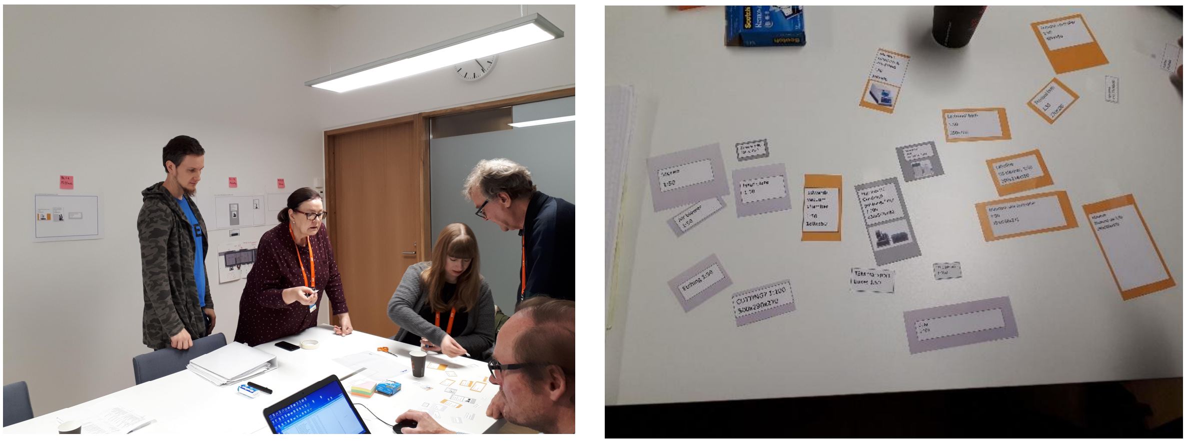


Figure 2.58. Planning the layout for microstructural characterisation equipment in hot cells.

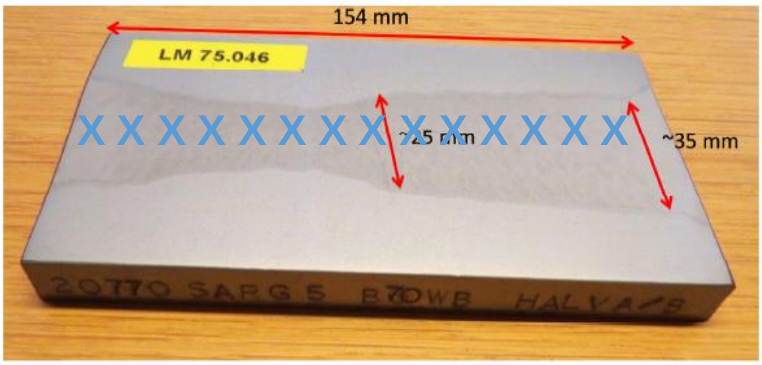


Figure 2.59. Appearance of a reference weld, showing schematically the locations of the chemical analysis planned.

The work in WP3, stakeholders, consists of networking and reporting progress with and to the Swedish BREDA (Barsebäck R&D arena) project and co-operation with Finnish stakeholders. An advisory board for the BREDA project has been established and meetings held on the progress in BRUTE2018 and preparation of the SAFIR2022 BRUTE proposal. A Finnish stakeholder group has been established and meetings held. Technical meetings have been frequent following the principle of well-prepared is half done.

The seminar “Reactor pressure vessel embrittlement seminar for the BREDA - BRUTE project” was very successful with high-class presentations from senior experts and doctoral students and with good discussions, and more than 30 participant from Finland and Sweden (Figure 2.60). BRUTE received publicity through an interviewed by Lännen Media.

Figure 2.60. More than 30 participants participated in the BRUTE RPV embrittlement seminar, and listened to talks, inclusing that by K. Lindgren, KTH, summarising her thesis work.

The WP4 includes the paper work needed by VTT for transportation of the samples to VTT. The first transportation of radioactive material was realised in the summer of 2018.

Deliverables in 2018

- Document describing the necessary steps needed to prepare, set-up and validate the mechanical testing methods to be used in BREDA.
- Final draft test matrix agreed on including options for more than 1500 specimens for future prioritisation and optimisation of available material and resources.
- Document describing the necessary steps needed to prepare, set-up and validate the microstructural characterization methods to be used.
- Totally 17 meetings with Finnish and Swedish stakeholders and project team held to disseminate the progress and plan for the execution of mechanical and microstructural investigation in SAFIR2022 BRUTE.
- Notes and presentations from the seminar distributed to SAFIR2018 and other stakeholders.
- First transportation of irradiated B2 materials successfully realised.

3. Financial and statistical information

The planned and actual volumes of the SAFIR2018 programme in 2018 were 7,12 M€ and 7,19 M€, and 45 and 49 person years, respectively. The funding partners were VYR with 4,339 M€, VTT with 1,480 M€, Lappeenranta University of Technology with 0,194 M€, NKS with 0,149 M€, Aalto University with 0,139 M€, Halden Reactor Project with 0,129 M€, TVO with 0,121 M€, Fennovoima with 0,103 M€, SSM with 0,103 M€, and other partners with 0,433 M€. The planned and actual funding by the major funding partners are illustrated in Figure 3.1. The planned and actual costs by cost category are shown in Figure 3.2. The personnel costs make up the major share.

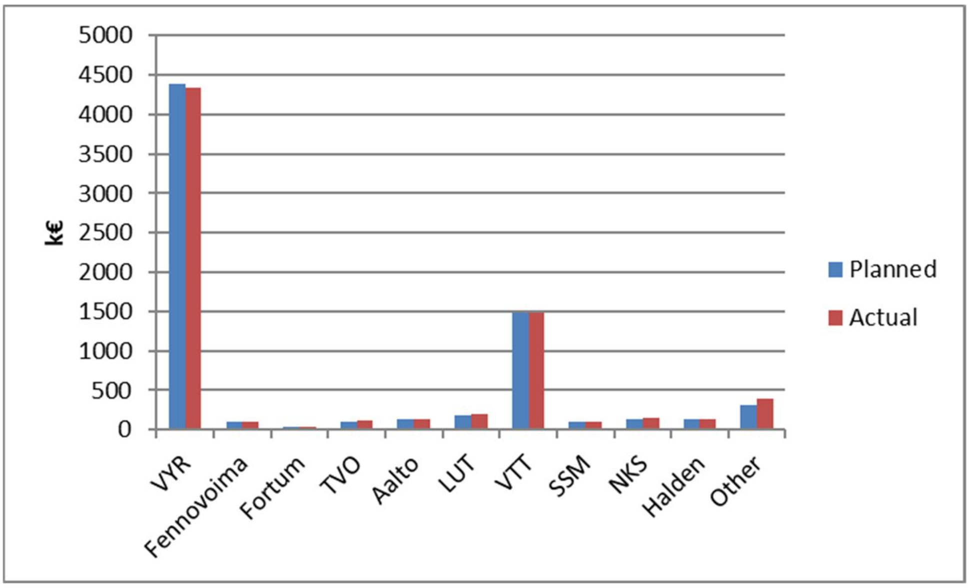


Figure 3.1. Planned and actual funding of the SAFIR2018 programme in 2018.

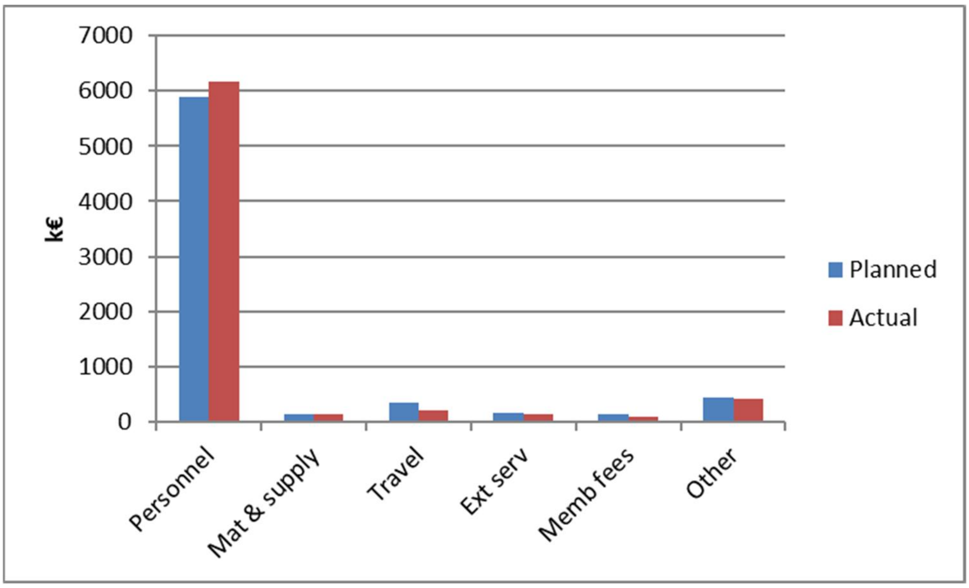


Figure 3.2. Planned and actual costs of the SAFIR2018 programme in 2018.

Figures 3.3-3.6 show the cost and volume distribution by research area. In the figures, the following abbreviations are used for the steering group research areas: SG1 Plant safety and systems engineering, SG2 Reactor safety, SG3 Structural safety and materials, and RG6 Research infrastructure.

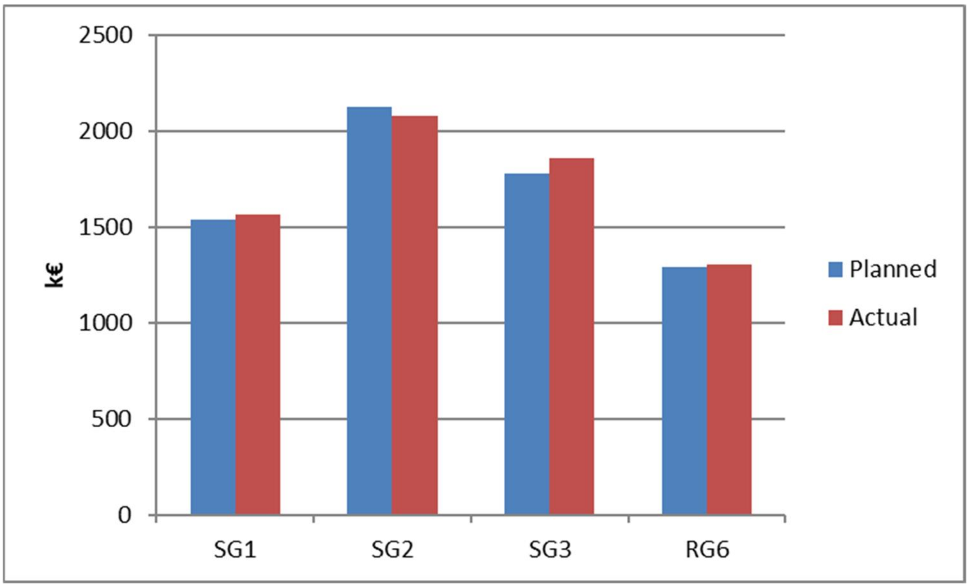


Figure 3.3. Planned and actual costs by research area in 2018.

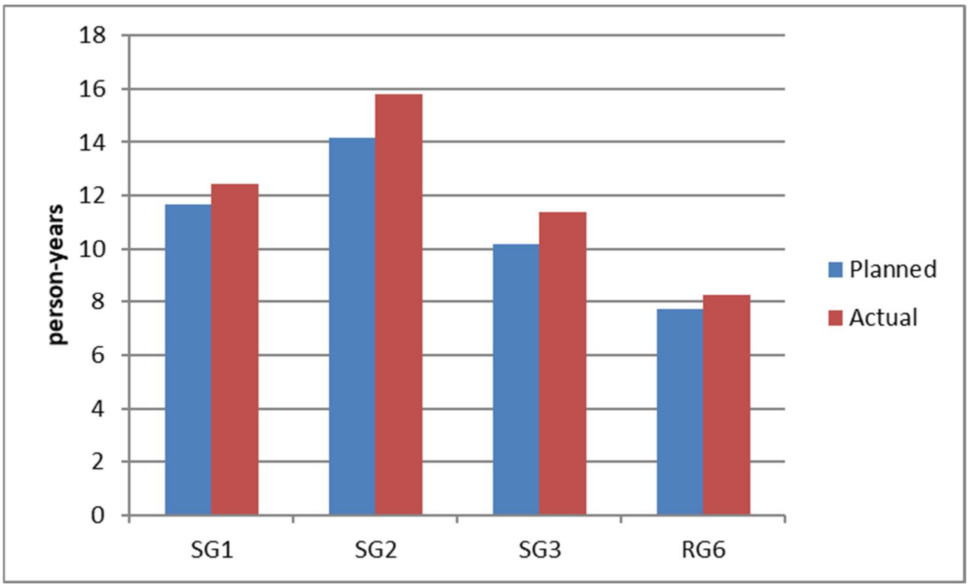


Figure 3.4. Planned and actual volumes by research area in 2018.

The actual costs coincided well with the planned costs in all research areas (Figure 3.3). On the other hand, the actual volumes in person-years were higher than the planned volumes (Figure 3.4). The fact is also reflected in the higher actual than planned personnel costs (Figure 3.2).

In the research areas SG3 and RG6 the shares of person-years were lower than their shares of total funding because of higher other than personnel costs (Figures 3.5-3.6). In SG1 and

SG2 the shares of the person-years were bigger than their shares of the total funding, respectively.

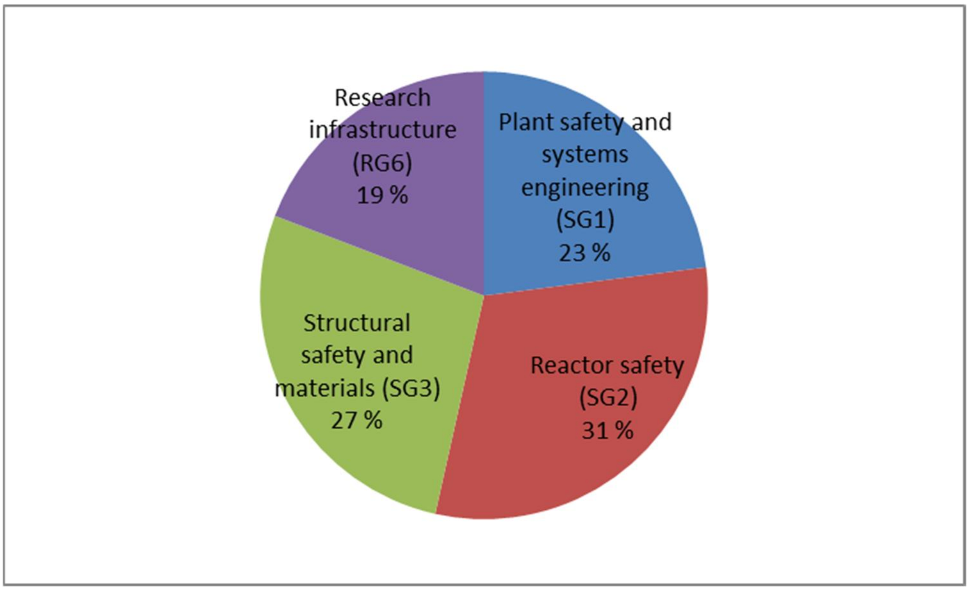


Figure 3.5. Distribution of total funding in SAFIR2018 research areas in 2018.

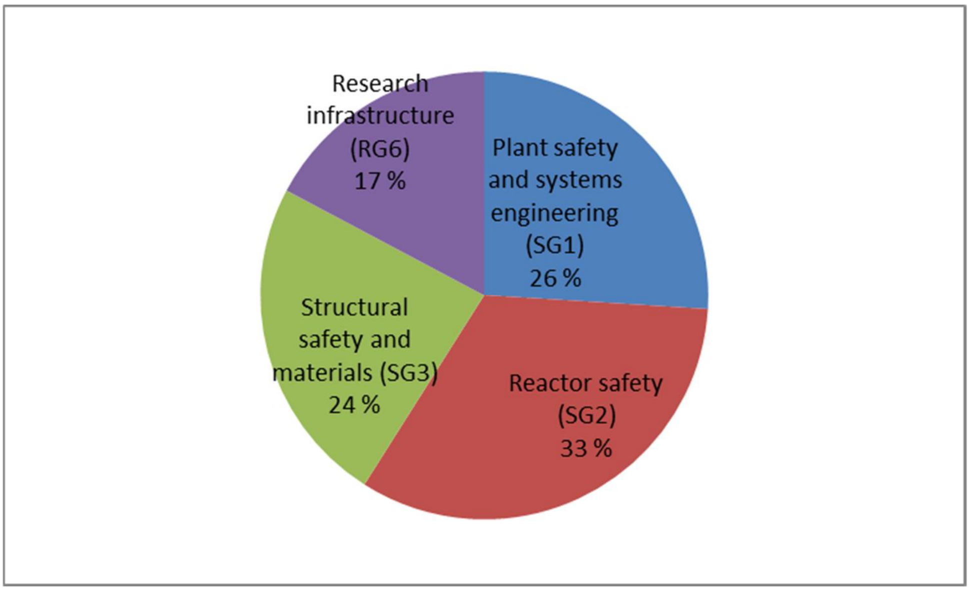


Figure 3.6. Distribution of person-years in SAFIR2018 research areas in 2018.

The numbers of different kind of publications made in SAFIR2018 research projects during 2018 are listed in Table 3.1. The programme produced 296 publications in 2018 consisting of 41 scientific journal articles, 60 conference articles, 141 research reports of the participating organisations, and 54 other publications (theses, reports of other organisations, etc.).

The average number of publications in the research projects was 6,1 per person-year, and the average number of scientific journal articles was 0,86 per person-year. There were differences in the number and type of publications between the projects the most of the projects wrote also scientific and conference articles in addition to research reports.

Table 3.1. Publications in the SAFIR2018 projects in 2018.

Project acronym	**Volume (person years)**	**Research reports**	**Scientific journal articles**	**Conference articles**	**Others**	**Total** (number of publications)
CORE	1,4	4	0	5	8	**17**
EXWE	2,6	5	6	0	2	**13**
MAPS	1,0	1	2	2	2	**7**
PRAMEA	2,4	11	3	3	3	**20**
SAUNA	2,8	2	1	9	2	**14**
GENXFIN	0,6	3	0	1	6	**10**
ESSI	1,1	5	1	4	1	**11**
ORSAPP	0,5	1	0	0	0	**1**
CASA	1,4	6	0	1	8	**15**
CATFIS	0,8	3	1	0	3	**7**
COVA	1,8	6	0	0	0	**6**
INSTAB	1,2	3	3	1	0	**7**
INTEGRA	3,4	5	2	0	1	**8**
KATVE	1,5	4	3	2	0	**9**
MONSOON	1,0	4	1	7	0	**12**
NURESA	1,4	5	2	0	0	**7**
PANCHO	1,9	5	0	1	1	**7**
SADE	0,7	4	1	0	0	**5**
USVA	0,8	4	0	0	0	**4**
ERNEST	0,6	1	3	0	0	**4**
FIRED	1,4	3	3	1	0	**7**
FOUND	1,9	9	1	4	1	**15**
LOST	1,8	6	1	2	2	**11**
MOCCA	0,8	1	1	1	1	**4**
THELMA	1,4	6	2	6	2	**16**
WANDA	1,3	6	1	5	2	**14**
COMRADE	0,7	3	1	2	2	**8**
EVOGY	1,5	1	1	1	0	**3**
INFRAL	2,1	3	1	2	0	**6**
JHR	0,2	0	0	0	3	**3**
RADLAB	4,3	12	0	0	4	**16**
BRUTE	1,6	6	0	0	0	**6**
ADMIRE	1,2	3	0	0	0	**3**
Total	**49,0**	**141**	**41**	**60**	**54**	**296**

Altogether 6 higher academic degrees were obtained in the research projects in 2018: one Doctoral degree and five Master's degrees (Table 3.2). The academic degrees are listed in Appendix 3.

Table 3.2. Academic degrees obtained in the projects in 2018.

Project acronym	**Doctor**	**Master**
EXWE		1
ESSI		1
INTEGRA		1
PANCHO	1	
EVOGY		1
INFRAL		1
Total	**1**	**5**

4. Programme management

The organisation of SAFIR2018 is shown in Figure 4.1 and its function described in detail in the Operational management handbook ([4], available on SAFIR2018 website).

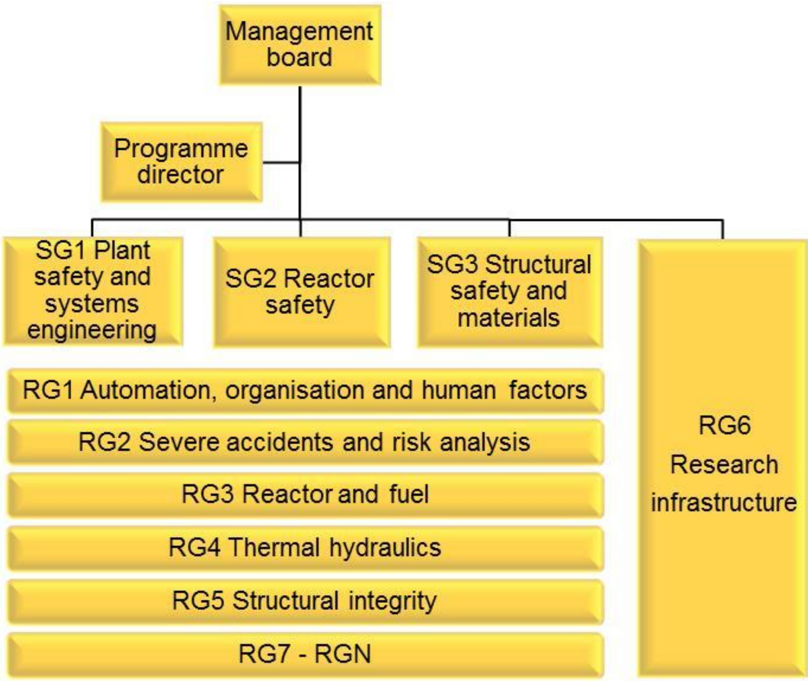


Figure 4.1. Structure of SAFIR2018 organisation. In 2018 each project belonged to one research area Steering Group (SG) and one Reference Group (RG1-RG6). RG6 "Research infrastructure" has a special role of a steering and reference group [4].

During the administrative period (January 2018 – March 2019) the SAFIR2018 Management Board (MB) held 4 meetings. Each of the steering groups SG1-SG3 had 3 meetings and RG6 as a steering group had also 3 meetings (the last meeting was held as an email meeting). The reference groups RG1-RG6 had 4 meetings (the last meetings were held as email meetings) and reference group RG6 had 3 meetings.

The SAFIR2018 management board has annually initiated small preliminary type studies with the order procedure. Decisions on the small projects are made after the funding recommendation for the research projects. The small projects support the implementation of the framework plan in topics where actual research projects have not been started. They can also introduce new topics. In 2018 one project was ordered and carried out: (1) Survey of 3D printing in nuclear industry (VTT). The project was formally realised as subcontracting in the administration project (ADMIRE). The final reports of the small projects carried out during the programme can be found on SAFIR2018 extranet.

A competence survey about the number of experts in different fields of nuclear energy sector in Finland was carried out in 2017. A report of the results was published both in Finnish [7] and in English [8].

The programme director participated in the work of the Euratom Programme Committee (Fission configuration) as an expert member (meeting in November 2018) and one meeting of the national support group was also organised by SAFIR2018. The programme director also participated in the work of OECD NEA CSNI (meeting in December 2018).

The information on the research performed in SAFIR2018 is communicated formally via the progress reports of the projects for the reference group meetings, the annual reports of the programme and SAFIR2018 website (public and protected extranet). Additional information was given in seminars organised by the research projects. The detailed scientific results were published as articles in scientific journals, conference papers, and research reports.

The final seminar of SAFIR2018 was held on 21.-22.3.2019 at Hanasaari. The seminar was also held as a webinar. The seminar material consisting of the programme and abstracts, slides and posters, and recorded video presentations can be found on SAFIR2018 public website (link). The SAFIR2018 Final report [6] describing the work carried out in the programme can also be found on the website.

References

1. National Nuclear Power Plant Safety Research 2015-2018. SAFIR2018 Framework Plan. Publications of the Ministry of Employment and the Economy, Energy and the climate 34/2014. (in Finnish, an English version also available on http://safir2018.vtt.fi/)

2. Hämäläinen, J. & Suolanen, V. (eds.) SAFIR2014 – The Finnish Research Programme on Nuclear Power Plant Safety 2011-2014. Final Report. VTT, Espoo, 2015. VTT Technology 213. 722 p. ISBN 978-951-38-8226-6; 978-951-38-8227-3 http://safir2014.vtt.fi/

3. Hämäläinen, J. & Suolanen, V. SAFIR2018 Annual Plan 2018. Research Report VTT-R-02884-18. 60 p + 722 app. http://safir2018.vtt.fi/

4. SAFIR2018 – The Finnish Research Programme for Nuclear Power Plant safety 2015-2018. Operational Management Handbook. 2015. 19 p + 26 app (available on http://safir2018.vtt.fi/).

5. Hämäläinen, J. & Suolanen, V. (eds.) SAFIR2018 – The Finnish Research Programme on Nuclear Power Plant Safety 2015-2018. Interim Report. VTT, Espoo, 2017. VTT Technology 294. 387 p. ISBN 978-951-38-8524-3. http://safir2018.vtt.fi/

6. Hämäläinen, J. & Suolanen, V. (eds.) SAFIR2018 – The Finnish Research Programme on Nuclear Power Plant Safety 2015-2018. Final Report. VTT, Espoo, 2019. VTT Technology 349. 498 p. ISBN 978-951-38-8682-0. http://safir2018.vtt.fi/

7. Hämäläinen, J. & Suolanen, V., Survey of Competence in the Nuclear Energy Sector 2017-2018 (Ydinenergia-alan osaamisselvitys 2017-2018; in Finnish). VTT Technology 344. Espoo 2018. 66 p + 6 app. ISBN 978-951-38-8676-9. https://www.vtt.fi/inf/pdf/technology/2018/T344.pdf

8. Hämäläinen, J. & Suolanen, V., Survey of Competence in the Nuclear Energy Sector 2017-2018 in Finland. Publications of the Ministry of Economic Affairs and Employment. Energy 22/2019. 85 p. ISBN: 978-952-327-410-5. http://urn.fi/URN:ISBN:978-952-327-410-5

Appendix 1

Publications in the projects in 2018

Crafting operational resilience in nuclear domain (CORE)

Conference articles and abstracts

Korpela, J., Hierarchical Gaussian modelling of instantaneous heart rate distributions. In: StanCon 2018, 29-31 August, Helsinki, Finland.

Laarni, J., Descriptive modelling of team troubleshooting in nuclear domain. In: Proceedings of the 12th Multi Conference on Computer Science and Information Systems (MCCSIS 2018), July 17-20, 2018, Madrid, Spain.

Laarni, J. Cognitive heuristics and biases in process control and maintenance work. In press to the Proceedings of the ANS NPIC&HMIT 2019.

Teperi, A-M., Applying human factors to promote a positive safety culture. Occupational and Environmental Medicine 75 (Suppl 2), 32nd Triennal Congress of the International Commission on Occupational Health (ICOH), 29 April – 4 May, 2018, Dublin, Ireland.

Teperi, A.-M., Human factors at safety critical fields. Tutorial. In: 9th AHFE International Conference, Florida, Orlando, USA, 21–25 July, 2018.

Research reports

Laarni, J., Supporting operational resilience. Final report. VTT: Espoo.

Liinasuo, M., & Koskinen, H., Principles and practises of emergency exercises. VTT: Espoo.

Pakarinen, S., & Korpela, J., WP4: Supporting operator performance in extreme stress. Final report. TTL: Helsinki.

Teperi, A-M, Puro, V., Tiikkaja, M., Ratilainen, H., 2018. Developing and implementing a human factors (HF) tool to improve safety management in the nuclear industry. Research report, available at: http://urn.fi/URN:ISBN 978-952-261-802-3.

Others

All researchers of the CORE project, Operatiivisen toimintavalmiuden kehittäminen ydinvoima-alalla. A slide set.

Pakarinen, S., Korpela, J., Karvonen, H., & Laarni, J., Modelling the cardiac indices of stress and performance of nuclear power plant operators during simulated fault scenarios. A journal article manuscript.

Puro, V., Kannisto, H., Lantto, E., & Teperi, A.-M., Learning from operating experiences in nuclear power plants – Organisational learning perspective. Accepted for publication. // Revised by SAFIR RG-member

Skjerve, A.B., Viitanen, K., Koskinen, H., Liinasuo, M., & Axelsson, C., Learning from successful operations in nuclear power plants – a guideline.

Teperi, A.-M., Gotcheva, N., & Aaltonen, K., Utilizing HF perspective to renew safety management in nuclear industry: design science perspective. Accepted for publication. // Revised by SAFIR RG-member

Wahlström, M., Seppänen, L, Heikkilä, H., & Kuula, T., Organizational and task-related considerations in creating reflective auto-confrontation – a seven-step checklist and two case studies on safety critical work. A journal article manuscript.

Viitanen, K., Integrative Sociotechnical Success Analysis Method with Case Example from Onagawa Nuclear Power Station. A journal article manuscript.

Viitanen, K., Integrating Learning from Successes into Operating Experience Process in the Nuclear Industry. A journal article manuscript.

Extreme weather and nuclear power plants (EXWE)

Scientific journal articles

Jylhä K, M. Kämäräinen, C. Fortelius, H. Gregow, J. Helander, O. Hyvärinen, M. Johansson, A. Karppinen, A. Korpinen, R. Kouznetsov, E. Kurzeneva, U. Leijala, A. Mäkelä, H. Pellikka, S. Saku, J. Sandberg, M. Sofiev, A. Vajda, A. Venäläinen, J. Vira, 2018: Recent meteorological and marine studies to support nuclear power plant safety in Finland. Energy, 165 (A), 1102-1118, https://doi.org/10.1016/j.energy.2018.09.033 (open access)

Kämäräinen M, Hyvärinen O, Vajda A, Nikulin G, Meijgaard E, Teichmann C, Jacob D, Gregow H, Jylhä K., 2018: Estimates of present-day and future climatologies of freezing rain in Europe based on CORDEX regional climate models. Journal of Geophysical Research: Atmospheres, 123. https://doi.org/10.1029/2018JD029131

Leijala U., Björkqvist J.-V., Johansson M.M., Pellikka H., Laakso L., Kahma K.K., 2018. Combining probability distributions of sea level variations and wave run-up to evaluate coastal flooding risks. Natural Hazards and Earth System Sciences, 18, 2785-2799. https://www.nat-hazards-earth-syst-sci.net/18/2785/2018/

Luomaranta, A., Aalto, J. and Jylhä, K., 2019: Snow cover trends in Finland over 1961-2014 based on gridded snow depth observations.. Int J Climatol. 2019;1–13. https://doi.org/10.1002/joc.6007

Olsson T, Post P, Rannat K, Keernik H, Perttula T, Luomaranta A, Jylhä K, Kivi R, Voormansik T, 2018: Sea-effect snowfall case in the Baltic Sea region analysed by reanalysis, remote sensing data and convection-permitting mesoscale modelling. Geophysica, 53(1), 65-91. http://www.geophysica.fi/pdf/geophysica_2018_53_olsson.pdf .

Pellikka, H.; Leijala, U.; Johansson, M. M.; Leinonen, K.; Kahma, K. K., 2018: Future probabilities of coastal floods in Finland. Continental Shelf Research, 157, 32–42. https://www.sciencedirect.com/science/article/pii/S0278434316302060

Conference abstracts

Lehtonen, I., Jylhä, K., 2018: Tendency towards more extreme precipitation climate in the CMIP5 models. EMS Annual Meeting Abstracts, 15, EMS2018-341. https://meetingorganizer.copernicus.org/EMS2018/EMS2018-341.pdf

Luomaranta, A., Aalto, J., Jylhä, K., 2018: Decreasing snow depth accompanied with mixed snowfall trends in Finland in 1961-2014. EMS Annual Meeting Abstracts, 15, EMS2018-284. https://meetingorganizer.copernicus.org/EMS2018/EMS2018-284.pdf

Toivonen E., J.-P. Pietikäinen, K. Jylhä, A. Mäkelä, H. Korhonen, 2018: Precipitation extremes and dry spells in the present and future climate in Finland modeled with a state-of-the-art high-resolution regional climate model. Geophysical Research Abstracts, 20, EGU2018-14981. https://meetingorganizer.copernicus.org/EGU2018/EGU2018-14981.pdf

Others

Hyvärinen, O., Vajda, A., 2018: Freezing rain and wind speed at the nuclear power plant sites in Finland. EXWE/SAFIR2018 deliverable D1.3.1 in 2018, Finnish Meteorological Institute.

Jylhä, K. 2018: Ilmasto muuttuu yhä nopeammin. Mitä Missä Milloin 2019, Vuosikirja syyskuu 2017 - elokuu 2018, 69. vuosikerta. Kustannusyhtiö Otava. ISBN: 9789511324881, p. 237-245.

Korpinen, A. 2018: Rajakerroksen simulointi heterogeenisellä alustalla. Master’s thesis, University of Helsinki, Institute of Atmospheric and Earth System Research, December 2018.

Mäkelä, A., Helander, J., Schulz, W., Haapalainen, J., Laapas, M., Räihä, J., Saku, S., Jylhä, K., 2018: Extreme value distribution of lightning peak current determined from lightning location data. EXWE/SAFIR2018 deliverable D1.1.3 in 2018, Finnish Meteorological Institute.

Olsson, T., Luomaranta, A., Jylhä, K., 2018: Detecting the atmospheric conditions favoring the development of sea-effect snowfall on the Finnish coast from HARMONIE simulations. . EXWE/SAFIR2018 deliverable D1.2.1 in 2018, Finnish Meteorological Institute.

Pellikka, H., Johansson, M., 2019: Mean sea level change at the Finnish NPP sites by 2100 under the RCP scenarios. EXWE/SAFIR2018 deliverable D2.2.2 in 2018, Finnish Meteorological Institute.

Pellikka, H., Johansson, M.M., Nordman, M., Bilker-Koivula, M., Ruosteenoja, K., 2019: Probabilistic projections for regional sea level rise in Finland by 2100. EXWE/SAFIR2018 deliverable D2.2.1 in 2018, Finnish Meteorological Institute.

Sofiev, M., Kouznetsov, R., Palamarchuk, Y., 2019: Assessment of the high-resolution SILAM dispersion & dose-assessment system. EXWE/SAFIR2018 deliverable D3.1.1 in 2018, Finnish Meteorological Institute.

Särkkä, J. and Johansson, M., 2018: Sea level extremes simulated from atmospheric reanalyses in the Baltic Sea. EXWE/SAFIR2018 deliverable D2.1.1 in 2018, Finnish Meteorological Institute.
Author(s): Jani Särkkä and Milla Johansson

Ukkonen, P. and Mäkelä, A., 2018a: Evaluation of machine learning models for predicting the initiation of deep convection. EXWE/SAFIR2018 deliverable D1.1.1 in 2018, Finnish Meteorological Institute.

Ukkonen, P. and Mäkelä, A., 2018b: Climate change impacts on summer convection in Finland. EXWE/SAFIR2018 deliverable D1.1.2 in 2018, Finnish Meteorological Institute.

Management Principles and Safety Culture in Complex Projects (MAPS)

Journal articles

Pargar, F., Kujala, J., Aaltonen, K. and Ruutu, S. (2019) Value creation dynamics in a project alliance, *International Journal of Project Management*. https://doi.org/10.1016/j.ijproman.2018.12.006

Aaltonen, K., Gotcheva, N., Kujala, J., Artto, K., Making sense of an innovation in a safety-critical megaproject. Under review in *International Journal of Managing Projects in Business.*

Book chapters

Gotcheva, N. (2019) Management Principles and Safety Culture in Complex Projects, in Holt, E., Kinnunen, P., Sevelev, D. (Eds.) *Impacts from VTT Research on Nuclear Safety and Radioactive Waste Management*. VTT Technical Research Centre of Finland, Espoo, Finland, 120 p. ISBN 978-951-38-8705-6.

Gotcheva, N., Aaltonen, K. and Kujala, J. (in press). Governance for safety in inter-organizational project networks, *Safety Science Research: Evolutions, Challenges and New Research Directions*, J-Ch. Le Coze (Ed.), Routledge.

Conference papers and presentations

Viitanen, K., Reiman, T., Rollenhagen, C. and Gotcheva, N. (2018) Mapping Methodical Change in Safety Culture, in Conference Proceedings of PSAM14, The Probabilistic Safety Assessment & Management, September 16-21, 2018, CA, Los Angeles, USA.

Pargar, F., Kujala, J., Aaltonen, K. and Ruutu, S. (2018) Value creation in project alliances: A qualitative system dynamics approach, presented at The International Research Network on Organizing by Projects (IRNOP) conference, RMIT University, Melbourne, Australia, 10-12 December 2018.

Research reports

Viitanen, K., Gotcheva, N., Rollenhagen, C. and Reiman, T. (2018) Safety Culture Assurance and Improvement Methods in Complex Projects – Final Report from the NKS-R SC_AIM, NKS-R, Organisational Issues and Safety Culture Research Area, NKS-405, ISBN 978-87-7893-493-2. http://www.nks.org/en/nks_reports/view_document.htm?id=111010214694225

Probabilistic risk assessment method development and applications (PRAMEA)

Scientific journal articles

A. Mancuso, M. Compare, A. Salo, E. Zio. Portfolio optimization of safety measures for the prevention of time-dependent accident scenarios. Conditionally accepted by Reliability Engineering and System Safety (2018).

A. Mancuso, P. Zebrowski, A. Couce Vieira. Risk-based Selection of Mitigation Strategies for Cybersecurity of Electric Power Systems. Submitted to IEEE Transactions on Dependable and Secure Computing (2019).

Markus Porthin, Terhi Kling, Marja Liinasuo. Effects of Digitalization of Nuclear Power Plant Control Rooms on Human Reliability Analysis – a Review (under review in Reliability Engineering and Safety Science)

Conference articles and abstracts

Ola Bäckström, Erik Cederhorn, Xuhong He, Jan-Erik Holmberg, Tero Tyrväinen, T. SITRON — Site risk assessment developed for Nordic countries. In: Proceedings of Probabilistic Safety Assessment and Management PSAM 14, September 2018, Los Angeles, CA. Paper 179 (2018).

Jan-Erik Holmberg, Ola Bäckström, Erik Cederhorn, Carl Sunde, Tero Tyrväinen. Site risk analysis for nuclear installations — Nordic method developments and pilot studies. In: S. Haugen et al. (eds.) Safety and Reliability – Safe Societies in a Changing World: Proceedings of ESREL 2018, June 17–21, 2018, Trondheim, Norway, Taylor & Francis Group, London, Pp. 1581−1588 (2018)

Jan-Erik Holmberg, Cilla Andersson, Stefan Authén, Ola Bäckström, Xuhong He, Per Hellström, Anders Karlsson, Tero Tyrväinen. Nordic Pilot Studies on Site Risk Analysis (SITRON Project) — Risk Metrics Developments. In: Proceedings of the International Workshop on Status of Site Level PSA Developments organised by OECD/NEA Working group RISK, July 18–20, 2018 Munich, Germany, Paper II-8 (2018).

Research reports

Terhi Kling, Marja Liinasuo. Human error analysis for a hybrid control room − A case study. VTT Research Report VTT-R-00143-19.

Ola Bäckström, Xuhong He, Jan-Erik Holmberg, Tero Tyrväinen, SITRON — WP2 — Method development. Site level risk assessment. Report 212634-R-001, Lloyd's Register Consulting, Sundbyberg (2019).

Jan-Erik Holmberg. SITRON — Pilot study Forsmark 1 and 2 2018. Report 14124_R010, Risk Pilot AB, Espoo (2018). (limited distribution)

Jan-Erik Holmberg, Stefan Authén, Ola Bäckström, Xuhong He, Kim Björkman, Tero Tyrväinen, Salvatore Massaiu. Site risk analysis for nuclear installations. Final Report from the NKS-R SITRON activity, NKS-419, Nordic nuclear safety research, Roskilde (2019).

Salvatore Massaiu. Decision-making during severe and multi-unit accidents: Technical Support Centers and Emergency Response Organizations at Nordic Countries, IFE/F-2018/189, Institute for Energy Technology, Halden (2019).

Tero Tyrväinen, Kim Björkman, SITRON - Site PSA model management, VTT-R-06885-18.

Tero Tyrväinen, Ilkka Karanta, Dynamic containment event tree modelling techniques and uncertainty analysis, VTT-R-06892-18.

Tero Tyrväinen, Time-dependent probabilistic risk assessment in FinPSA, VTT-R-04119-18.

Teemu Mätäsniemi, Tero Tyrväinen, FinPSA - Software testing - Verification and validation report for time-dependent task file, VTT-R-00084-19.

Ilkka Karanta, Seasonal and context factors in level 3 PRA - survey of status in Finland, VTT-R-00142-19.

Kim Björkman, FinPSA - Software testing - Verification and validation report for risk integrator updates, VTT-R-00014-19.

Others

Jan-Erik Holmberg. Travel report on IAEA Consultancy Meeting on the development of the Safety Report on Human Reliability Assessment for Nuclear Installations, Vienna, 19−22 February 2018, Report 14124_R009, Risk Pilot AB, Espoo (2018).

Jan-Erik Holmberg. SITRON (SITe Risk Of Nuclear installation) stakeholders meeting, April 16, 2018, notes, Report 14124_R011, Risk Pilot AB, Espoo (2018).

Ilkka Karanta, PRAMEA ad hoc meeting, June 5, 2018, minutes, VTT, Espoo 2018.

Integrated safety assessment and justification of nuclear power plant automation (SAUNA)

Scientific journal articles

Bergroth, J., Koskinen, H., Laarni, J. Use of Immersive 3-D Virtual Reality Environments in Control Room Validations, Nuclear Technology, Volume 202, Issue 2-3, 2018, pp. 278-289. https://dx.doi.org/10.1080/00295450.2017.1420335

Conference articles and abstracts

Buzhinsky, I., Pakonen, A., Vyatkin, V. Synthesis-aided reliability assurance of basic block models for model checking purposes, 27th IEEE International Symposium on Industrial Electronics (ISIE 2018), Cairns, Australia, June 13–15, 2018, pp. 669–674.
https://doi.org/10.1109/ISIE.2018.8433793

O'Halloran, B. M., Papakonstantinou, N., Van Bossuyt, D. L., Assessing the Consequence of Cyber and Physical Malicious Attacks in Complex, Cyber-Physical Systems During Early System Design, 16th IEEE International Conference on Industrial Informatics (INDIN 2018), Porto, Portugal, July 18–20, 2018, pp. 733–740.
https://doi.org/10.1109/INDIN.2018.8471937

Holmberg, J.-E. Risk-informed safety classification of component of auxiliary systems for emergency diesel generators in nuclear power plants. European Safety and Reliability Conference ESREL 2018, Trondheim, Norway, June 17–21, 2018, pp. 1589–1597.

Linnosmaa, J., Valkonen, J., Karpati, P., Hauge, A., Sechi, F., Gran, B. A.. Towards model-based specification and safety assurance of nuclear I&C systems – Applicability of SySML and AADL, 11th International Topical Meeting on Nuclear Plant Instrumentation, Control and Human-Machine Interface Technologies (NPIC & HMIT 2019), Orlando, FL, February 9–14, 2019.

Pakonen, A., Buzhinsky, I. Verification of fault tolerant safety I&C systems using model checking. 20th IEEE International Conference on Industrial Technology (ICIT 2019), Melbourne, Australia, February 13–15, 2019, pp. 969–974.

Pakonen, A., Buzhinsky, I., Vyatkin, V. Counterexample visualization and explanation for function block diagrams, 16th IEEE International Conference on Industrial Informatics (INDIN 2018), Porto, Portugal, July 18–20, 2018, pp. 747–753.
https://doi.org/10.1109/INDIN.2018.8472025

Papakonstantinou, N., Linnosmaa, J., Alanen, J., O'Halloran, B., Automatic fault tree generation from multidisciplinary dependency models for early failure propagation assessment, ASME 2018 International Design Engineering Technical Conferences & Computers and Information in Engineering Conference (IDETC/CIE 2018), Quebec City, Canada, August 26–29, 2018.
https://doi.org/10.1115/DETC2018-85189

Papakonstantinou, N., Linnosmaa, J., Alanen, J., Bashir, A. Z., O'Halloran, B., Van Bossuyt, D. L., Early hybrid safety and security risk assessment based on interdisciplinary dependency models, 65th Annual Reliability and Maintainability Symposium (RAMS 2019), Orlando, FL, January 28–31, 2019.

Varkoi, T., Mäkinen, T., Cameron, F., Nevalainen, R. Validating effectiveness of compliance evaluation in safety-critical process assessment, 25th European Conference on Systems, Software and Services Process Improvement (EuroSPI 2018), Bilbao, Spain, September 5–7, 2018, pp. 181–192.
https://doi.org/10.1007/978-3-319-97925-0_15

Research reports

Buzhinsky, I. Assessment of formal nuclear I&C logic models using behavior examples. Technical report, Aalto University (2018).
https://doi.org/10.13140/RG.2.2.26467.20009

Nevalainen, R., Varkoi, T. Nuclear SPICE Handbook 1.0, FiSMA report 2018-1.

Tyrväinen, T., Porthin, M. Probabilistic risk model of digital reactor protection system, VTT Research Report VTT-R-06631-18.

Others

Koskinen, H., Laarni, J. Practical method for training simulator fidelity evaluation, slides.

Holmberg, J.-E. SAUNA project meeting “Risk-informed decision making in safety engineering” 2018-11-09 notes, Report 14127_R003, Risk Pilot AB, Espoo, 2018.

Safety of new reactor technologies (GENXFIN)

Conference articles and abstracts

Tulkki, V., Pursiheimo, E., Lindroos, T.J., Sahlberg, V., Small Modular Reactor as a Part of a District Heating System. Transactions of the American Nuclear Society, Philadelphia, Pennsylvania, June 17–21, 2018, pp. 685–687. http://ansannual.org/wp-content/2018/Data/pdfs/333-24569.pdf

Research reports

Ilvonen, M., Considerations in EPZ sizing of SMR plants, VTT Research Report VTT-R-06998-18.

Sevón, T., Modeling of PANDA isolation condenser experiments with MELCOR, VTT Research Report, VTT-R-00021-19.

Ylätalo, J., Fortum SMR Modelling 2018, Fortum Report.

Others

Penttilä, S., Travel report from the 12th Information Exchange Meeting on Supercritical Water Cooled Reactors R&D, Montreal, Canada, March 19–20, 2018.

Penttilä, S., Travel report from the Generation IV International Forum Supercritical Water Cooled Reactors Materials & Chemistry Meeting, Madrid, Spain, September 6–7, 2018.

Penttilä, S., Travel report from the International Conference on Generation IV and Small Reactors, Ottawa, Canada, November 6–8, 2018.

Penttilä, S., Travel report from the IAEA Coordinated Research Project Meeting on Accident Tolerant Fuels, Vienna, Austria, November 26–30, 2018.

Tulkki, V., Travel report from the IAEA Technical Meeting on the Deployment of Non-Electric Applications Using Nuclear Energy for Climate Change Mitigation, Vienna, Austria, April 16–18, 2018.

Tulkki, V., Travel report from the American Nuclear Society Annual Meeting, Philadelphia, Pennsylvania, USA, June 17–21, 2018.

Electric Systems and Safety in Finnish NPP (ESSI)

Scientific journal articles

Rizk, M.E.M., Lehtonen, M., Baba, Y., Abulanwar, S. Performance of Large-Scale Grounding Systems in Thermal Power Plants Against Lightning Strikes to Nearby Transmission Towers, IEEE Transactions on Electromagnetic Compatibility, Volume: 61, Issue: 2, April 2019, pp. 400-408. https://doi.org/10.1109/TEMC.2018.2831700

Conference articles and abstracts

Alahäivälä, A., Lehtonen, M. Analysis of open phase condition influence on an induction motor, 2018 19th International Scientific Conference on Electric Power Engineering, EPE 2018, Brno, Czech Republic, 16-18 May 2018, Proceedings, pp. 1-6. https://doi.org/10.1109/EPE.2018.8395990

Subedi, D., Lehtonen, M. Lightning Overvoltages in Electrical Power System of a Power Plant. Submitted to 20th International Scientific Conference on Electric Power Engineering, EPE 2019, Kouty nad Desnou, Czech Republic, 15-17 May 2019. 7 p.

Lehtonen, M., Pichler, M., Schürhuber, R., Ground Potential Rise and Lightning Overvoltages in Control Systems of Large Power-Plants under High Soil Resistivity. Submitted to 20th International Scientific Conference on Electric Power Engineering, EPE 2019, Kouty nad Desnou, Czech Republic, 15-17 May 2019, 4 p.

Kulmala, A., Alahäivälä, A. 2018. Open Phase Condition Scenarios for Nuclear Power Plant Electrical Network Studies. Submitted to 13th IEEE PES PowerTech Conference, 23-27 June 2019, Milan, Italy.6 p.

Research reports

Holmberg, J.-E. 2018, Preliminary risk analysis for adaptive operation of NPP. Espoo. Risk Pilot Report 17141_R001. 13 p.

Pasonen, R. 2018. Risks of adaptive control of NPP electrical systems and stability of the grid. Espoo: VTT. VTT Research report VTT-R-04700-18. 15 p. https://www.vtt.fi/inf/julkaisut/muut/2018/VTT-R-04700-18.pdf

Pasonen, R. 2018. Power plant lightning overvoltage protection of low voltage power electronics. Espoo: VTT. VTT Research report VTT-R-06945-18. 21 p. https://cris.vtt.fi/ws/portalfiles/portal/22508321/VTT_R_06945_18.pdf

Kulmala, A. 2019. Unbalances caused by different OPC cases in NPP electric systems and preparedness of operating plants against OPC. Espoo: VTT. VTT Research report VTT-R-000019-19. 18 p.

Kulmala, A., Alahäivälä, A. 2019. Methods for detection of the OPC condition and recommendations to improve safety of NPPs in the case of OPC. Espoo: VTT. VTT Research report VTT-R-000020-19. 22 p.

Others

Gürbüz, I.T., Lightning Induced Over-voltages in Nuclear Power Plants. Master´s thesis. Aalto School of Electrical Engineering, 2018. pp. 82+8 appendixes. https://aaltodoc.aalto.fi/handle/123456789/35755

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

Research reports

Ylönen, M., ORSAPP, Institutional strength-in-depth in the context of decommissioning and learning from incidents, VTT-R-00139-19.

Comprehensive Analysis of Severe Accidents (CASA)

Conference papers

Sevón, Tuomo. 2018. Severe Accident Analyses for the License Application Review of Olkiluoto 3 EPR. CSARP meeting, June 5–7, 2018, Rockville, Maryland.

Anna Nieminen, 2018. Analysis of WH-24 experiment with ASTEC. OECD/NEA THAI-5 PRG5 meeting. October 1–2, 2018, Frankfurt, Germany.

A. Bentaib, A., Chaumeix, N., Grosseuvres, R., Bleyer, A., Gastaldo, L., Maas, L., Jallais, S., Vyazmina, E., Kudriakov, S., Studer, E., Dehbi, A., Halouane, Y., Schramm, B., Taivassalo, V., Frankova, M., Kotsuba, O., Holler, T., Kljenak, I., Maruyama, Y., Nuri, T., Sato, M., Murgatroyd, J., Povilaitis, M., 2018. ETSON-MITHYGENE benchmark on simulations of upward flame propagation experiment in the ENACCEF2 experimental facility. 12th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Operation and Safety (NUTHOS-12). Qingdao, China, October 14-18, 2018.

Research institute reports

Ilvonen, Mikko & Rossi, Jukka. 2019. Comparison of MACCS, ARANO and VALMA. VTT Research Report VTT-R-00136-19. 18 p. + app. 9 p.

Korpinen, Anna, Strandberg, Magnus, Sevón, Tuomo, Kärkelä, Teemu & Hokkinen, Jouni. 2019. Retention of fission products in a containment pool - Parts A and B. VTT Research Report VTT-R-00152-19. 43 p. + app. 12 p.

Sevón, Tuomo. 2019. Testing of MELCOR’s New Core Catcher Model. VTT Research Report VTT-R-00061-19. 8 p.

Strandberg, Magnus. 2019. Simulating passive containment cooling with MELCOR. VTT Research Report VTT-R-00146-19. 14 p.

Taivassalo, Veikko & Strandberg, Magnus. 2019. Blind simulation of a hydrogen deflagration test in the two-vessel THAI+ facility. VTT Research Report VTT-R-00181-19. 59 p.

Taivassalo, Veikko. 2019. Comparison of simulations for debris bed behaviour in post-dryout conditions. VTT Research Report VTT-R-00223-19. 47 p.

Others

Sevón, Tuomo. Travel report from the CSARP/MCAP meeting. 4 p.

Chemistry and transport of fission products (CATFIS)

Scientific publications

Gouëllo, M., Kalilainen, J., Kärkelä, T., Auvinen, A., Contribution to the understanding of iodine transport under primary circuit conditions: CsI/Cd and CsI/Ag interactions in condensed phase, Submitted to Nuclear Materials & Energy (2018).

Research institute reports

Gouëllo, M., Hokkinen, J., Experimental study on the behaviour of CsI on primary circuit surfaces of a nuclear power plant: Effects of molybdenum on iodine transport: part II, VTT-R-04457-18 (2018).

A combined report of two sub projects in CATFIS: Kärkelä, T., Gouëllo, M., Penttilä, K., Kekki, T., Kotiluoto, P., HNO_3 formation by beta radiation and the effect on sump pH – Research report in 2018, VTT-R-00023-19 (2019). (Confidential)

A combined report of CATFIS and CASA projects: Part A (Experiments) - Kärkelä, T., Hokkinen, J. (CATFIS), Part B (Simulations) - Korpinen, A., Strandberg, M., Sevon, T. (CASA), Retention of fission products in a containment pool - Parts A and B, VTT-R-00152-19 (2019).

Others

Travel account of OECD/NEA STEM-2 meeting, Paris, France, June 2018. (Confidential)

Travel account of OECD/NEA BIP-3 meeting, Paris, France, June 2018. (Confidential)

Conference paper: Gouëllo, M., Hokkinen, J., Kärkelä, T., Experimental study of the effect of molybdenum on iodine transport in the primary circuit during a severe nuclear power plant accident, In proceedings of International Congress on Advances in Nuclear Power Plants (ICAPP 2018), Charlotte, USA, April 8-11, 2018.

Comprehensive and systematic validation of independent safety analysis tools (COVA)

Research institute reports

Szogradi, M. 2018. Validation of Apros 6.08 with Becker's burnout experiments. Espoo, VTT. VTT-R-05067-18

Airaksinen, R. 2018. RBHT Reflooding Heat Transfer Experiments with Apros. Espoo, VTT. VTT-R-03931-18

Silde, A. 2019. Calculation of TOSQAN spray test 101 using the multi-size spray droplet model of the Apros containment code. Espoo, VTT. VTT-R-00081-19

Szogradi, M. 2018. Analysis of PKL4 i2.2 run3 benchmark with Apros 6.07. Espoo, VTT. VTT-R-02244-18.

Kolehmainen, J. 2018. Apros simulations of the PASI characterizing experiments. Espoo, VTT. VTT-R-04602-18

Hillberg, S. 2019. BNL critical flow experiments with Apros. Espoo, VTT. VTT-R-05798-18

Couplings and instabilities in reactor systems (INSTAB):

Scientific publications

Gallego-Marcos, I., Kudinov, P., Villanueva, W., Kapulla, R., Paranjape, S., Paladino, D., Laine, J., Puustinen, M., Räsänen, A., Pyy, L., Kotro, E., Pool stratification and mixing induced by steam injection through spargers; analysis of the PPOOLEX and PANDA experiments. Nuclear Engineering and Design 337 (2018), pp. 300-316. https://doi.org/10.1016/j.nucengdes.2018.07.004.

Hujala, E., Tanskanen, V., Hyvärinen, J., Frequency analysis of chugging condensation in pressure suppression pool system with pattern recognition. Nuclear Engineering and Design 339 (2018), pp. 224-252. https://doi.org/10.1016/j.nucengdes.2018.09.018.

Gallego-Marcos, I., Kudinov, P., Villanueva, W., Kapulla, R., Paranjape, S., Paladino, D., Laine, J., Puustinen, M., Räsänen, A., Pyy, L., Kotro, E., Pool stratification and mixing induced by steam injection through spargers: CFD modelling of the PPOOLEX and PANDA experiments. Submitted to NED for review on 12 June 2018, revision requested on 14 January 2019.

Conference papers and abstracts

Pättikangas, T., Hovi, V., Puustinen, M., Räsänen, A., Kotro, E., CFD simulations of direct-contact condensation of horizontal vapor jets. Abstract accepted to: The 18th International Topical Meeting on Nuclear Reactor thermal Hydraulics (NURETH-18), Portland, Oregon, USA, August 18-23, 2019 (Log Number: 27710).

Research institute reports

Puustinen, M., Laine, J., Räsänen, A., Kotro, E., Tielinen, K., SEF-POOL Tests. Lappeenranta-Lahti University of Technology LUT. School of Energy Systems. Nuclear Engineering. Research Report INSTAB 1/2018. Lappeenranta 2019. 29 p. + app. 4+3 p.

Puustinen, M., Laine, J., Räsänen, A., Kotro, E., Tielinen, K., Erosion of Thermal Stratification in PPOOLEX Test with Sparger. Lappeenranta-Lahti University of Technology LUT. School of Energy Systems. Nuclear Engineering. Research Report INSTAB 2/2018. Lappeenranta 2019. 20 p. + app. 14 p.

Patel, G., Literature review of non-condensible dissolution / release dynamics studies and their effects in reactor coolant circuit. Lappeenranta-Lahti University of Technology LUT. School of Energy Systems. Nuclear Engineering. Research Report INSTAB 3/2018. Lappeenranta 2019. 20 p.

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

Scientific publications

Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, Juhani Hyvärinen, Experimental observation of adverse and beneficial effects of nitrogen on reactor core cooling, Nuclear Engineering and De-sign, vol. 332 (2018), Elsevier, pp. 111-118, https://doi.org/10.1016/j.nucengdes.2018.03.027.

Otso-Pekka Kauppinen, Virpi Kouhia, Vesa Riikonen, Juhani Hyvärinen, System code analysis of accumulator nitrogen discharge during LOCA experiment at PWR PACTEL test facility, has been offered for publishing to Nuclear Engineering and Design on 4.2.2019.

Research institute reports

Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, PWR PACTEL pump trip experiments, Research report, INTEGRA 1/2018, Lappeenranta University of Technology / Nuclear Engineering, Lappeenranta, 2018, 8 + 4 s.

Vesa Riikonen, Quick look report of the PWR PACTEL nitrogen reference experiment in the OECD/NEA PKL Phase 4 project, Research report, INTEGRA 2/2018, Lappeenranta University of Technology / Nuclear Engineering, Lappeenranta, 2018, 10 + 10 s.

Virpi Kouhia, Vesa Riikonen, PASI facility – characterizing experiments, Research report, INTEGRA 3/2018, Lappeenranta University of Technology / Nuclear Engineering, Lappeenranta, 2018, 43 + 7 s.

Virpi Kouhia, Simulation of PWR PACTEL Pump Trip Experiment with APROS code, Research report, INTEGRA 4/2018, Lappeenranta University of Technology / Nuclear Engineering, Lappeenranta, 2018, 24 + 1 s.

Virpi Kouhia, Vesa Riikonen, Joonas Telkkä, Harri Partanen, Antti Räsänen, Otso-Pekka Kauppinen, Eetu Kotro, Ilkka Saure, Kimmo Tielinen, Juha Karppinen, General description of the PASI test facility, second edition, Research report, INTEGRA 5/2018, Lappeenranta-Lahti University of Technology LUT / Nuclear Engineering, Lappeenranta, 2018, 22 + 32 s.

Others

Ilkka Tahvola, Modelling of PKL test facility with TRACE code, master's thesis, Lappeenranta University of Technology, 2018.

Nuclear criticality and safety analyses preparedness at VTT (KATVE)

Scientific journal articles

Arkoma, A., Huhtanen, R., Leppänen, J., Peltola, J. and Pättikangas, T., "Calculation chain for the analysis of spent nuclear fuel in long-term interim dry storage" Annals of Nuclear Energy, Vol. 119, p. 129 – 138 https://doi.org/10.1016/j.anucene.2018.04.037

Kaltiaisenaho, T., "Photon transport physics in Serpent 2 Monte Carlo code", manuscript submitted to Radiation Physics and Chemistry

Leppänen, J., "Response Matrix Method Based Importance Solver and Variance Reduction Scheme in the Serpent 2 Monte Carlo Code", manuscript submitted to Nuclear Technology

Conference articles and abstracts

Kaltiaisenaho, T. "Photonuclear reactions in Serpent 2 Monte Carlo code", extended summary submitted to The International Conference on Mathematics and Computational Methods applied to Nuclear Science and Engineering (M&C 2019), Portland, Oregon, USA, August 25-29, 2019.

Leppänen, J. and Jokipii, M. "Global variance reduction scheme with self-adaptive weight-window mesh in the Serpent 2 Monte Carlo code", extended summary submitted to The International Conference on Mathematics and Computational Methods applied to Nuclear Science and Engineering (M&C 2019), Portland, Oregon, USA, August 25-29, 2019.

Research reports

Arkoma, A., "Modelling cladding hydrides in interim dry storage", VTT Research report, VTT-R-05604-18

Dorval, E., "Generation of problem-dependent, traceable dosimetry libraries for reactor fluence spectral adjustment", VTT Research report, VTT-R-04159-18

Häkkinen, S., "Calculations of a burnup credit benchmark #ISTC 2670 VVER PIE", VTT Research report, VTT-R-00144-19

Juutilainen, P. and Valtavirta, V., "Status of the criticality safety validation package in 2018", VTT Research report, VTT-R-00228-19

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

Scientific journal articles

Sahlberg, V. "Recalculating the steady state conditions of the V-1000 zero-power facility at Kurchatov Institute using Monte Carlo and nodal diffusion codes." Kerntechnik, 82 (2018) 420-425. https://doi.org/10.3139/124.110810

Conference articles and abstracts

Sahlberg, V. and Rintala, A. "Preliminary Verification of the Rectangular Diffusion Solver of Ants." In proc. PHYSOR 2018, Cancun, Mexico, Apr. 22-26, 2018.

Valtavirta, V. and Leppänen, J. "Effects of Fuel Temperature History Approximations on Neutronics Properties of a BWR Assembly." In proc. PHYSOR 2018, Cancun, Mexico, Apr. 22-26, 2018.

Valtavirta, V. and Leppänen, J. "New Stochastic Substep Based Burnup Scheme for Serpent 2." In proc. PHYSOR 2018, Cancun, Mexico, Apr. 22-26, 2018.

Kuopanportti, J., Saarinen, S. and Leppänen, J. "Assessment of Serpent Cross Sections of the Second Generation Fuel of TVEL for Loviisa NPP." In proc. PHYSOR 2018, Cancun, Mexico, Apr. 22-26, 2018

Rintala, A. and Sahlberg, V. "Extension of Nodal Diffusion Solver of Ants to Hexagonal Geometry." In Proc. 28th Symposium of AER on VVER Reactor Physics and Reactor Safety, October 8 – 12, 2018, Olomouc, Czech Republic.

Suikkanen, H. and Rintala, V. “Assessing the Performance of Functional Expansion Tallies in the Serpent 2 Monte Carlo Code.” Abstract submitted to International conference M&C 2019.

Valtavirta, V., Hovi, V., Loukusa, H., Rintala, A., Sahlberg, V., Tuominen, R. and Leppänen, J. “Kraken - an upcoming Finnish reactor analysis framework.” Abstract submitted to International conference M&C 2019.

Research reports

Leppänen, J. "Neutron cross section libraries for Serpent 2 – Preliminary release (Version 0)." VTT-R-00118-18, VTT Technical Research Centre of Finland, Ltd., 2018.

Kaltiaisenaho, T. “Photon, atomic relaxation and electron data library for Serpent 2, version 0.1.” VTT-R-00154-19, VTT Technical Research Centre of Finland, 2019.

Suikkanen, H. and Rintala, V. “Description of the coupled Serpent - TRANSURANUS calculation system and performance tests of the function expansion power tally.” Lappeenranta University of Technology, 2018.

Rintala, A. Implementation of HEXBU-3D/MOD5 group constant model in Ants.” VTT-R-03315-18, VTT Technical Research Centre of Finland, 2018.

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

T. Pättikangas and V. Hovi, CFD calculations of the condensation in the small-scale separate effect facility. (VTT).

T. Pättikangas and V. Hovi, CFD calculations of PPOOLEX sparger tests. (VTT).

J. Peltola, Updated documentation of the reactingEulerFoam solvers in thermal phase change simulations. (VTT).

J. Peltola, Polydisperse phase change simulations with OpenFOAM. (VTT).

G. Patel and V. Tanskanen, single orifice sparger and collar blowdown pipe simulations. (LUT).

V. Hovi, Coupled Apros and CFD models for the simulation of PWR pressurizers. (VTT).

T. Rämä, Behavior of pressurizer during loss-of-feedwater transient of VVER-440. (Fortum).

Physics and Chemistry of Nuclear Fuel (PANCHO)

Conference articles and abstracts

Loukusa, H., Valtavirta, V., Application of constrained Gibbs energy minimization in nuclear fuel thermochemistry. TopFuel2018, 30 September-4 October 2018, Prague, Czechia.

Research reports

Loukusa, H., Valtavirta, V. FINIX - Fuel behavior model and interface for multiphysics applications - Code documentation for version 1.18.9, Research report, VTT-R-06824-18, Espoo, 2018.

Peltonen, J. FINIX - Fuel behavior model and interface for multiphysics applications - Validation of version 1.18.9, Research report, VTT-R-06823-18, Espoo, 2018.

Loukusa, H., Peltonen, J., Valtavirta, V. FINIX - Fuel behavior model and interface for multiphysics applications - Code documentation for version 1.19.1, Research report, VTT-R-00052-19, Espoo, 2019.

Loukusa, H. Modeling missing pellet surface defects with BISON, Research report, VTT-R-00067-19, Espoo, 2019.

Arkoma, A. WGFS RIA Fuel-Code Benchmark Phase 3 – SCANAIR uncertainty and sensitivity studies on CABRI CIP0-1 test, Research report, VTT-R-00149-19, Espoo, 2019.

Others

Heikinheimo, J., Pakarinen, J., Bes, R., Makkonen, I., Tuomisto, F. Insight into irradiation-induxed high-temperature vacancy clusters in cerium dioxide. Journal article draft.

Safety analyses for dynamical events (SADE)

Research institute reports

Räty H. "Coupling the fuel behaviour module FINIX-0.17.12 to the three-dimensional reactor dynamics code HEXTRAN 4.0", Research report VTT-R- 06735-18

Sahlberg V. “Changelog and report on notable changes made to the HEXTRAN code for heterogeneous fuel modelling”. Research report VTT-R-00045-19

Syrjälahti E., “Kalinin-3 -- Model report and simulation of the benchmark problem”. Research report VTT-R-06914-18

Syrjälahti E., “HEXTRAN-SMABRE development and validation of the internally coupled simulation mode in 2018”. Research report VTT-R-06913-18

Uncertainty and sensitivity analyses for reactor safety (USVA)

Research reports

Arkoma, A., Sensitivity analysis of global uncertainties in large break loss-of-coolant accident (LB-LOCA) simulations, VTT Research Report, VTT-R-00137-19.

Szógrádi, M., Methodology for determining input uncertainties of Apros, VTT Research Report, VTT-R-04148-18.

Valtavirta, V., Nuclear data uncertainty propagation to Serpent generated group and time constants, VTT Research Report, VTT-R-04681-18.

Syrjälahti, E., Preliminary simulations of the Kalinin-3 benchmark problem for uncertainty and sensitivity analyses, VTT Research Report, VTT-R-06972-18.

Experimental and numerical methods for external event assessment improving safety (ERNEST)

Scientific journal articles

Saarenheimo, A., Borgerhoff, M., Calonius, K., Darraba, A., Hamelin, A., Ghadimi Khasraghy, S., ... Välikangas, P. (2018). Numerical studies on vibration propagation and damping test V1. *Rakenteiden Mekaniikka*, *51*(1), 55 - 80. https://doi.org/10.23998/rm.68954

Fedoroff, A.,, Calonius, K., Kuutti, J., Behavior of the Abaqus CDP model in simple stress states. To be published in Rakenteiden Mekaniikka (Journal of Structural Mechanics)

Fedoroff, A.,, Calonius, K., Using the Abaqus CDP model in impact simulations. To be published in Rakenteiden Mekaniikka (Journal of Structural Mechanics)

Research reports

Vepsä,A. Two punching resistance tests of plain concrete slabs with different maximum aggregate sizes, VTT Research Report, VTT-R-00040-19.

FIre Risk Evaluation and Defence-in-Depth (FIRED)

Scientific journal articles

Jukka Vaari, Antti Paajanen. “Evaluation of the reactive molecular dynamics method for research on flame retardants: ATH-filled polyethylene.” Computational Materials Science 153 (2018) 103-112. https://doi.org/10.1016/j.commatsci.2018.06.032

Teemu Isojärvi, Hadi Bordbar, Simo Hostikka. “Spectrally resolved calculation of thermal radiation penetration into liquid n-heptane in pool fires.” International Journal of Heat and Mass Transfer 127 (2018) 1101-1109. https://doi.org/10.1016/j.ijheatmasstransfer.2018.08.108

Deepak Paudel, Simo Hostikka. “Propagation of modelling uncertainty in stochastic heat-transfer simulation using a chain of deterministic models". International Journal for Uncertainty Quantification. Accepted for publication 12.12.2018.
http://www.dl.begellhouse.com/journals/52034eb04b657aea,forthcoming,27275.html
https://doi.org/10.1615/Int.J.UncertaintyQuantification.2018027275

Conference articles and abstracts

Jukka Vaari, Antti Paajanen. “Molecular simulation of fire retardancy: solid-phase effects of aluminium hydroxide.” 22nd International Symposium an Analytical and Applied Pyrolysis (Pyro2018), Kyoto, Japan June 3-8, 2018.

Research reports

Jukka Vaari: Kinetic parameters for a continuum-scale pyrolysis model from atomistic simulations. VTT-R-06948-18.

Topi Sikanen: Model based optimization for calibration of pyrolysis models. VTT-R-05578-18.

Deepak Paudel, Simo Hostikka. Modeling the fire resistance test of stone-wool based fire barriers. Aalto report 27.12.2018, 24p.

Other

The PyroPython software is available online at https://github.com/PyroId/PyroPython.
The PyroPython documentation is available at http://pyroid.github.io/

Analysis of fatigue and other cumulative ageing to extend lifetime (FOUND)

Scientific journal articles

Kuutti, J. and Virkkunen, I. 2019. The effect of subsurface crack opening on the stress intensity factor under cyclic thermal loads. Manuscript submitted to Engineering Fracture Mechanics 13.2.2019.

Conference articles and abstracts

Seppänen, T. 2018. Low cycle fatigue of AISI 304L and 347 in PWR water. Proceedings of the ASME 2018 Pressure Vessels and Piping Conference PVP2018. July 15-20, Prague, Czech Republic

Timperi, A. 2018a. MODELLING OF FATIGUE DUE TO THERMAL MIXING BY USING A SPECTRUM METHOD. Proceedings of the ASME 2018 Pressure Vessels and Piping Conference PVP2018. July 15-20, Prague, Czech Republic

Kuutti, J. 2018. Comparison of ASME XI and BS7910 allowable surface flaw size evaluation procedures in piping components. Proceedings of the ASME 2018 Pressure Vessels and Piping Conference PVP2018. July 15-20, Prague, Czech Republic

Cronvall, O. 2018. Long-term operation of BWR RPV and its internals. Proceedings of the ASME 2018 Pressure Vessels and Piping Conference PVP2018. July 15-20, Prague, Czech Republic

Research reports

Keinänen, H. and Oinonen, A. 2019. Stress and degradation analyses of the pump deck weld. VTT Research Report VTT-R-00036-19. Espoo: VTT.

Seppänen, T., 2018. Fatigue of AISI 347 and 304L Stainless Steels in PWR Water. VTT Research Report VTT-R-06961-18. Espoo: VTT.

Oinonen, A. 2018. Probabilistic Leak-Before-Break Analysis for a Pipe Elbow. VTT Research Report VTT-R-03118-18. Espoo: VTT.

Oinonen, A. 2018. Fluid-structure-interaction analysis for a water hammer loaded piping elbow. VTT Research Report VTT-R-02253-18. Espoo: VTT.

Tyrväinen, T. 2018. Computation of consequence measures of piping component failures. VTT Re-search Report VTT-R-02261-18. Espoo: VTT

Timperi, A. 2018. Study on Temperature Spectra for Thermal Fatigue Modelling. VTT Research Report VTT-R-06881-18. Espoo: VTT.

Timperi, A. 2018. Study of B2 Stress Indices for Elbows. VTT Research Report VTT-R-06882-18. Espoo: VTT.

Timperi, A. 2019. Study on Probabilistic Combination of Pipe Moments. VTT Research Report VTT-R-00028-19. Espoo: VTT.

Virkkunen 2019. Dissimilar metal weld residual stress measurements using the contour method. Aalto University, Research report, Revision 1, 29.1.2017.

Others

Cronvall, O. 2018. SUSCEPTIBILITY OF BOILING WATER REACTOR PRESSURE VESSEL AND ITS INTERNALS TO DEGRADATION. Doctoral Dissertation, Manuscript draft, 2018. Aalto University.

Long term operation aspects on structural integrity (LOST)

Scientific journal articles

Lindqvist, S., A crack-location correction for T0 analysis of an Alloy 52 dissimilar metal weld, Engineering fracture mechanics, Espoo, 2018.

Conference articles and abstracts

Kim Wallin, Assessment of Master Curve material inhomogeneity using small data sets, Pressure vessel and piping conference, 2018.

Lindqvist, S., Seppänen, T., Characterization of J-R curves of a HSLA-steel and an Alloy 52 DMW with SE(T) specimens, European Conference on Fracture, Espoo, 2018.

Research reports

Heikki Keinänen, Weld repair residual stress simulations and comparison to experimental measurements, VTT-R-01487-18, Espoo, 2018.

Sebastian Lindqvist, Heikki Keinänen, Applicability of HRR stress field ahead of a crack at an interface between a hard and a soft material, VTT-R-06951-18, Espoo, 2018

Saifi, Q. 2018. Development of special numerical approach for dam-age mechanical analysis with implementation in Abaqus. Research Report. Espoo: VTT. VTT Technology VTT-R-00018-19. 23 p.

Lindqvist, S. 2018. Pre-investigation for BREDA−various aspects related to aging of the reactor pressure vessel. Espoo. VTT-R-06941-18. 29 p.

Lindqvist, S. Boåsen, M. Constraint and fracture toughness. 2018. VTT- R-06942-18. Espoo.

Lindqvist, S., Sirkiä, L., Karjalainen-Roikonen, P. 2018. Tearing resistance under isothermic and decreasing thermal transients. Espoo. VTT-R-06958-18. 17 p.

Others

Pentti Arffman, Mechanical properties of DMWs, Master’s Thesis, Espoo, 2019.

Sebastian Lindqvist, Travel report from ASTM conference, Travel report, Espoo, 2018.

Mitigation of cracking through advanced water chemistry (MOCCA)

Scientific publications

Bojinov, M., Jäppinen, E., Saario, T., Sipilä, K. & Toivonen, A. 2019. ”Effect of lead and applied potential on corrosion of carbon steel in acidic and alkaline steam generator crevice solutions”. 27 p. Presented for publication in Corrosion Science journal.

Conference papers

Sipilä, K., Ikäläinen, T. & Saario, T., Presentation: “Application of Controlled-Distance Electrochemistry-Arrangement on Electrochemical Impedance Spectroscopy Measurements in Low Conducting Electrolyte”. 17th Nordic Corrosion Congress, 23-25 May 2018, Technical University of Denmark.

Research institute reports

Betova, I., Bojinov, M. & Saario, T. 2018. Hydrazine alternatives as oxygen scavengers and passivation agents in steam generator preservation conditions. VTT Research Report VTT-R-01908-18, 35 p.

Others

Sipilä, K., Huotilainen, C., Saario, T., Peltonen, S. & Ikäläinen, T., Presentation: “On-line EIS measurements in simulated steam generator conditions”, 14th Annual Meeting of ECG-COMON, June 10-12, 2018, ZAG, Ljubljana, Slovenia.

Thermal ageing and EAC research for plant life management (THELMA)

Scientific journal articles

Huotilainen, C., Ivanchenko, M., Ehrnstén, U., Efsing, P. Electrochemical investigation of in-service thermal ageing and G-phase precipitation of two CF8M cast stainless steels. Submitted to Journal of Nuclear Materials.

Mouginot, R et al. Nanoindentation study on short-range ordering. Materials Research Express January 2019. https://iopscience.iop.org/article/10.1088/2053-1591/aafeaf/meta

Conference articles and abstracts

Ehrnstén, U., Carpén, L., Tompuri, K. Microbially induced corrosion in fire fighting systems – experience and remedies. The Minerals, Metals & Materials Society 2018. J.H. Jackson et al. (eds.), Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, The Minerals, Metals & Materials Series. https://doi.org/10.1007/978-3-319-67244-1_21

Mouginot, R. et al. Development of short-range order and intergranular carbide precipitation in Alloy 690 TT upon thermal ageing. The Minerals, Metals & Materials Society 2018. J.H. Jackson et al. (eds.), Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, The Minerals, Metals & Materials Series. https://doi.org/10.1007/978-3-319-67244-1_21

Ivanchenko, M. Microstructural investigations of the 300-series austenitic stainless steel irradiated to different fluences after in-pile mechanical test. Fontevraud 9 conference, 17-20 September 2018, Avignon, France.

Karlsen T., Hata, K., Ivanchenko M. In-pile crack growth rate studies on irradiated 304L and CW316 stainless steels under PWR conditions. Fontevraud 9 conference, 17-20 September 2018, Avignon, France.

Hurley, C., Yagodzinskyy, Y., Mouginot, R., Malitckii, E., Hänninen, H. Combined effect of thermal ageing and hydrogen. 3rd international conference on metals & hydrogen. 29-31.5.2018, Ghent, Belgium.

Hänninen, H. On the mechanisms of EAC of Alloys 690 and 52 in pressurized water reactor applications. Presentation at the 2018 Eurocorr conference, September, 9-13, 2018, ICE Krakow, Poland.

Research reports

Ahonen, M. Literature survey on high-Ni alloys used in VVER reactors. VTT-R-06912-18, 14.12.2018, 9p.

Ehrnstén, U. Progress report on EU-MEACTOS project in 2018. VTT-R-06699-18, 29.11.2018, 12 p.

Huotilainen, C. Progress report on EU-INCEFA+ for SAFIR2018 - THELMA. VTT-R- 06855-18,

Ivanchenko, M. Analytical electron microscopy of unirradiated boron alloyed stainless steel absorber sleeve material. VTT-R- 06938-18.

Lydman, J., Lindqvist, S. Nevasmaa, P and Ehrnstén, U. Microstructural characterization of non-irradiated Barsebäck RPV base material and weld metal. VTT research report VTT-R05125-18, 11.2.2019. 154 p.

Tapper, U. Microstructural characterization of non-irradiated Barsebäck RPV material: Transmission Electron Microscopy study. VTT-R-00041-19, 11.1.2019, 16p.

Others

Ahonen, M. Travel report from EPRI 690/152/52 research collaboration meeting, 13.12.2018, 5p.

Huotilainen, C et al. Effect of hydrogen and short-range ordering in thermally treated Alloy 690. Presentation at the ICG-EAC meeting, April 2018, Knoxville, US.

NDE of NPP primary circuit components and concrete infrastructure (WANDA)

Scientific journal articles

Koskinen, T. Virkkunen, I. Papula, S. Sarikka, T. Haapalainen, J. Producing a POD curve with emulated signal response data. Insight - Non-Destructive Testing and Condition Monitoring, Volume 60, Number 1, January 2018, pp. 42-48(7). https://doi.org/10.1784/insi.2018.60.1.42 (Sent for acceptance in 2017).

Conference articles and abstracts

Koskinen, T. Virkkunen, I. Hit/Miss POD with Model Assisted and Emulated Flaws. 12th European Conference on Non-Destructive Testing (ECNDT 2018), Gothenburg, Sweden, June 11–15, 2018.

Fahim Al-Neshawy, Paksuseinäisen betonirakenteen kuntotutkimus yhdistetyllä ainetta rikkomattomalla menetelmällä - case study. Betonitutkimusseminaari 2018, 31.10.2018, Helsinki.

Al-Neshawy F., Ferreira, M., Ojala, T. Sjöblom, V. and Puttonen, J. Conceptual design, construction and NDE of NPP reinforced concrete mock-up wall. Abstract: SMiRT 25, Charlotte, NC, USA, August 4-9, 2019.

Al-Neshawy F., Ferreira, M., Ojala, T. and Puttonen, J. Mock-up wall for non-destructive testing and evaluation of thick-reinforced concrete structures in nuclear power plants. 12th European Conference of Non-Destructive Testing in Gothenburg, Sweden, June 11 - 15, 2018

Ferreira, M., Bohner, E., Sjöblom, V., Al-Neshawy, F., Ojala, T., (2018) Nuclear Power Plant containment wall mock-up for long-term NDE. SMT and NDT-CE 2018, New Brunswick, NJ, USA, 27-29 August 2018

Ferreira, M., Bohner, E., Sjöblom, V., Al-Neshawy, F., Ojala, T., (2019) Construction of realistic NPP containment wall mock-up for challenging NDE methods. FIB Symposium 2019, Krakow, Poland (paper accepted)

Al-Neshawy, F., Ferreira, M., Ojala, T., Sjöblom, V., Puttonen, J., (2019) Conceptual design, construction and NDE of NPP reinforced concrete mock-up wall. SMiRT 25 Charlotte, NC, USA, August 4-9, 2019 (paper submitted)

Research reports

Rinta-Aho, J. Non-contact measurement method for Poisson's ratio, VTT Research Report, VTT-R-06820-18.

Ferreira, M. Sjöblom, V. Bohner, E. Al-Neshawy, F. Ojala, T. Final Report for the construction of a Non-destructive Evaluation mock-up, VTT Research Report, VTT-R-02769-18.

Al-Neshawy, F. Ojala, T. Ferreira, M. Sjöblom, V. Evaluation and calibration of NDT/NDE methods, VTT Research Report, VTT-R-06833-18.

2017 delayed reports:

Al-Neshawy, F. Huuskonen-Snicker, E. Ojala, T. Condition Assessment of Thick-Walled Concrete Structure -TVO Wall. Aalto University Technical report.

Ferreira, M. Anttila, V. Current status on the use of POD (Probability of Detection) for NDE of reinforced concrete structures. VTT Research Report, VTT-R-02133-18.

Ferreira, M. Sjöblom, V. Al-Neshawy, F. Ojala, T. Pre-construction of the thick-walled reinforced concrete mock-up - Design report. VTT Research Report, VTT-R-02092-18.

Others

Jessen-Juhler, O., Artificial Flaw Simulation in Ultrasonic Inspection of Austenitic Stainless Steel Weld. Master's thesis, Tampere University of Technology, September 2018. https://dspace.cc.tut.fi/dpub/handle/123456789/26445

Leskelä, E., PIONIC 2 meeting, Travel report.

Condition Monitoring, Thermal and Radiation Degradation of Polymers Inside NPP Containments (COMRADE)

Scientific journal articles

Paajanen, A., Vaari, J. and Verho, T. Crystallization of cross-linked polyethylene in molecular dynamics simulation. A manuscript has been submitted to Polymer journal.

Conference articles and abstracts

Molander, M., Jansson, A., Sandström, J. Development of condition monitoring methods for polymeric components including low dose rate radiation exposure. FONTEVRAUD 9 17–20 September 2018, Avignon, France.

Jansson, A., Carlred, L., Sjövall, P., Sipilä, K., Joki, H. Surface sensitive analysis for investigation of oxidation profiles in rubber sealing material. FONTEVRAUD 9 17–20 September 2018, Avignon, France.

Research reports

Simulation of rubber materials for seals in use in nuclear power plants (RISE-research report)

Title of the publication: Study of aged polymers from nuclear power plants (RISE-research report)

Ageing of EPDM and Lipalon cable materials in high temperature and irradiation environments (VTT-research report)

Others

Workshops on each power plant to discuss COMRADE results and how to use them (minutes of the meetings)

COMRADE Workshop on polymer ageing issues at NPPs (Minutes of the meeting)

Evolving the Fennoscandian GMPEs (EVOGY)

Scientific journal articles

Ludovic Fülöp, Vilho Jussila, Riina Aapasuo, Tommi Vuorinen, Päivi Mäntyniemi: Evolving the Fennoscandian GMPEs, manuscript in preparation.

Conference articles and abstracts

Vilho Jussila, Billy Fälth, Peter Voss, Björn Lund, Ludovic Fülöp: Synthetic ground motions to support the near-field seismic hazard prediction in Fennoscandia, Poster presentation in NKS seminar 15-16 January 2019

Research reports

Ludovic Fülöp, Vilho Jussila, Billy Fälth, Peter Voss, Björn Lund: Synthetic ground motions to support the near-field seismic hazard prediction in Fennoscandia, Research report to NKS, Final draft ready. In review in STUK.

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Scientific journal articles

Hujala, E., Tanskanen, V., Hyvärinen, J. Frequency analysis of chugging condensation in pressure suppression pool system with pattern recognition. Nuclear Engineering and Design 339 (244-252), 2018. https://doi.org/10.1016/j.nucengdes.2018.09.018

Conference articles and abstracts

Telkkä, J., Jämsén S., Hyvärinen, J. Defining Void Fraction with Axial Wire-mesh Sensor in a Swirling Two-phase Flow. Abstract accepted to: Specialists Workshop on Advanced Instrumentation and Measurement Techniques for Experiments Related to Nuclear Reactor Thermal Hydraulics and Severe Accidents (SWINTH-2019), Livorno, Italy, October 22–25, 2019.

Hujala, E., Tanskanen, V., Patel, G., Hyvärinen, J. Analysis of Bubbling Mode Condensation Oscillation in Horizontal Sparger. Abstract accepted to: 18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-18), Portland, Oregon, USA, August 18–23, 2019.

Research reports

Joonas Telkkä, Juha Karppinen, INFRAL 1/2018: Modular Test Loop (MOTEL), Research report, LUT University, Nuclear Engineering, 2018.

Joonas Telkkä, INFRAL 2/2018: Short report on PWR PACTEL/PACTEL maintenance, Research report, LUT University, Nuclear Engineering, 2018.

Joonas Telkkä, Elina Hujala, Lauri Pyy, INFRAL 3/2018: Report on advanced measuring techniques 2018, Research report, LUT University, Nuclear Engineering, 2018.

JHR collaboration & Melodie follow-up (JHR):

Others

Travel reports

- WG meeting 9.-10.10.2018
- Technical seminar 21.-22.3.2018
- OECD/NEA 2nd Workshop 4.-5.10.2018

Radiological laboratory commissioning (RADLAB)

Research reports

Karlsen, W., Moilanen, P., Functionality report of in-cell Electric Discharge Machine, VTT Report VTT-R-06874-18, 2018, 20p.

Karlsen, W., Jokipii, M., Functionality report of in-cell Electron Beam Welder, VTT Report VTT-R-06637-18, 2018, 13p.

Arffman, P., Functionality report of RKP450 impact test device with semi-automatic tempering and specimen feed, VTT Report VTT-R-06917-18, 2018, 13p.

Tähtinen, S., Saarinen, J., Arffman, P. Functionality report of in-cell 250kN material testing machine, VTT Report VTT-R-05557-18, 2018, 26p.

Tähtinen, S., Functionality report of the in-cell fatigue precracking devices, VTT Report VTT-R-00048-19, 2019, 20p.

Karlsen, W., Jokipii, M., Functionality report of in-cell optical dimensioning microscope, VTT Report VTT-R-06640-18, 2018, 18p.

Karlsen, W., Mattila, M., Functionality report of the in-cell hardness testing device, VTT Report VTT-R-06639-18, 2018, 8p.

Leskinen, A., Functionality report of ICP-OES Device, VTT Report VTT-R-04334-18, 2018, 76p.

Toivonen, A., Jokipii, M., Väisänen, P., Functionality report of hot autoclaves, VTT Report VTT-R-03555-18, 2018, 12p.

Palosuo, I., Arffman, P., Karlsen, W., VTT:n Ydinturvallisuustalon paikallissuojattavat testauslaitteet, VTT Report VTT-R-06971-17, 2018, 12 p.

Siivinen, J., VTT:n Ydinturvallisuustalon jätteiden käsittely - laitteiston toimintakuvaus, VTT Report VTT-R-06831-18, 2018, 17p.

Heikola, T., Lavonen, T., Myllykylä, E. Pätevyyskokeisiin osallistuminen-RADLAB, VTT Report VTT-R-03984-18, 2018, 9p.

Others

Heikola, T., Lavonen, T., Myllykylä, E., Quality Assurance Measurements with Reference Materials, Poster presentation 9th Nordic Conference on Plasma Spectrochemistry, June 9 - 13, 2018, Loen Norway

Lavonen, T., Heikola, T., Myllykylä, E., ELEMENTAL CHARACTERIZATION ANALYSIS OF DECOMMISSIONING MATERIALS FROM FiR1 TRIGA MARK II TYPE RESEARCH REACTOR, Poster presentation 9th Nordic Conference on Plasma Spectrochemistry, June 9 - 13, 2018, Loen Norway

Lavonen, T., ja Myllykylä, E., Travel report from the 9th Nordic Conference on Plasma Spectrochemistry, June 9 - 13, 2018, Loen Norway

Heikola, T., Travel report from the HR-ICP-MS (High Resolution Inductively Coupled Plasma Mass Spectrometry) Element 2 -training course, September 3-7, 2018, Bremen, Germany.

Barsebäck RPV material used for true evaluation of embrittlement (BRUTE)

Research reports

Boåsen, M. et al. Preliminary mechanical test matrix for the BREDA/NRUTE project. VTT Research Report VTT-R-06849-18, 10.01.2019, 24 p.

Ehrnstén, U. Notes from the BREDA-BRUTE SAFIR2018 seminar on pressure vessel embrittlement at VTT CNS, November 31st, 2018. 3 p.

Ehrnstén, U. Evaluation of surface requirement for hardness measurements in BRUTE. VTT Research Report VTT-R-06879-18, 12.12.2018, 13 p.

Lydman, J. Characterization plan for Barsebäck 2 trepan – BRUTE 2018. Deliverable 2.1. VTT Research report VTT R-01423-18, 6.4.2018, 19 p.

Lydman, J. Identification of initial microstructural heterogeneities promoting brittle fracture in RPV though SEM, FIB and investigations: visit to University of Manchester, Material Performance Centre 9.-13.7.2018. Report to Nugenia, 4p.

Planman, T. Background and overview of mechanical testing methods to be used in BREDA. VTT Research Report VTT-R-02227-18, 25.5.2018, 13 p.

Appendix 2

Participation in international projects and networks in 2018

Crafting operational resilience in nuclear domain (CORE)

OECD/NEA WGHOF (Working Group on Human and Organisational Factors) (Laarni, J.)

OECD/NEA WGHOF Task group on Achieving Reasonable Confidence in Validation Test Results of Integrated System Performance for Nuclear Power Plant Main Control Rooms (Laarni, J.)

NUGENIA (Nugenia Generation II & III Association) (Laarni, J.)

Member of Scientific Advisory Board, Co-chair in 'Human factors in Training, Education and Learning Sciences' session (TELS), Co-Editor of Conference Proceedings: Nazir S., Teperi A-M., Polak-Sopińska A. (eds) Advances in Human Factors in Training, Education, and Learning Sciences. AHFE 2018. Advances in Intelligent Systems and Computing, Vol. 785. Springer, Cham. https://doi.org/10.1007/978-3-319-93882-0_44 . 9th International Conference on Applied Human Factors and Ergonomics (AHFE 2018, 21–25 July, Orlando, Florida, USA) (Teperi, A.-M.)

Extreme weather and nuclear power plants (EXWE):

EU-C3S project DECM (Data Evaluation for Climate Models), co-designing recommendations for evaluation of multi-model climate model products CMIP and CORDEX (2016-2018)

EU-C3S project Clim4Energy, a service providing climate change indicators tailored for the energy sector (2016-2018)

ERA4CS project SERV_FORFIRE, Integrated services and approaches for assessing effects of climate change and extreme events for fire and post fire risk prevention (2017-2020)

EAR4CS project URCLIM, impacts of heat stress, flooding risk, and snow clearing and slipperiness in winter in alternative future urban environments (2017-2020)

ERA4CS project WINDSURFER, wind and wave scenarios, uncertainty and climate risk assessment for forestry, energy and reinsurance (2017-2020)

ESSEM COST Action ES1404 "A European network for a harmonised monitoring of snow for the benefit of climate change scenarios, hydrology and numerical weather prediction" (2014-201z)

COST-CA15211 Atmospheric Electricity Network: coupling with the Earth System, climate and biological systems. Finnish MC delegate from FMI.

NORDRESS is a Centre of Excellence under the Social Security Programme of NordForsk to carry out multidisciplinary studies to enhance societal security and resilience to natural disasters (2015-2019)

EUMETSAT Optical Lightning Imager (LI) Mission Advisory Group (MAG). FMI is an Invited Expert Institute.

Finnish Management Committee member, COST CA15211 “Atmospheric Electricity”, since 2016. http://www.cost.eu/COST_Actions/ca/CA15211

Nepal World Bank Project 2014-2018, Nepal. Funded by World Bank. Lightning location data usability in severe weather monitoring and early warning services.

FISU II ICI: Promoting Adaptation to Climate Change by Reducing Weather and Climate-Related Losses through Improved Services in Sudan and South Sudan, 2016-2018.

SSWSS: Severe Storm Warning Service for Sri Lanka, 2016-2018. Supporting the early warning services for thunderstorms in Sri Lanka. Funded by the Ministry of Foreign Affairs of Finland and the Finnish Agency for Technology and Innovation (Tekes).

Nordic Framework Climate Services (NFCS), Heavy Rainfall Activity, Finnish delegate, since 2016.

The international HIRLAM programme develops short range numerical weather prediction since the 1980's. http://hirlam.org/index.php/hirlam-programme-53

FMI is active in the following ECRA (European Climate Research Alliance) Collaborative Programmes: i) Arctic Climate Stability and Change; ii) High Impact Events and Climate Change; iii) Sea Level Change and Coastal Impacts

Collaboration with the European Centre for Medium-Range Weather Forecasts (ECMWF) for various subjects, including running case study simulations, and storm prediction development with (ERF) extended range forecasts.

Collaboration with Nansen Environmental and Remote Sensing Center (NERSC) for studies about extreme wave and sea level events.

Management Principles and Safety Culture in Complex Projects (MAPS)

Co-operation with KTH Royal Institute of Technology in Stockholm and the Swedish nuclear industry (Forsmark, OKG)

Co-operation with NKS - sub-project Safety Culture Assurance and Improvement Methods in Complex Projects (SC_AIM)

Co-operation with International Atomic Energy Agency (IAEA), Operational Safety Section, Division of Nuclear Installation Safety, Department of Nuclear Safety and Security in terms of safety culture continuous improvement.

Co-operation with INERIS, France (Jean-Christophe Le Coze) for an edited book *Safety Science Research: Evolutions, Challenges and New Research Directions*, Routledge.

Co-operation with RMIT University, Melbourne, Australia in the field of project organizing and project governance/alliance projects.

Probabilistic risk assessment method development and applications (PRAMEA):

NKS project SPARC (Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics) (Ilkka Karanta, Tero Tyrväinen)

NKS and Nordic co-operation project SITRON (SITe Risk Of Nuclear installations) (Jan-Erik Holmberg, Stefan Authén, Kim Björkman, Tero Tyrväinen)

ESREL2018 conference, Trondheim, June 17–21 2018

- Conference paper and presentation: Holmberg, J.-E., Bäckström, O., Cederhorn, E., Sunde, C., Tyrväinen, T. 2018a. Site risk analysis for nuclear installations — Nordic method developments and pilot studies.

FinPSA End User Group

OECD/NEA WGRISK (Working Group on Risk Assessment)

- Participation in group activities (Markus Porthin)
- Participation in the activities on Status of Site Level PSA (Including Multi-unit PSA developments) (Markus Porthin)
- International workshop on status of site level PSA (including multi-unit PSA) developments, Munich, Germany, July 18–20, 2018.

ETSON (European Technical Safety Organizations Network), working group on Probabilistic Safety Assessment

- Participation in group activities by workshop, meeting and email (Markus Porthin)

IAEA coordinated research project “Probabilistic Safety Assessment (PSA) Benchmark for Multi-Unit/Multi-Reactor Sites” (I31031)

IAEA expert group for the development of the Safety Report on Human Reliability Assessment for Nuclear Installations

- Member of the expert group (Jan-Erik Holmberg)

International Research on Disaster Risk (IRDR) research programme, Ahti Salo participates in follow-up

IIASA collaboration, Alessandro Mancuso spent three months at the International Institute for Applied Systems Analysis (IIASA) working on a project for Probabilistic Risk Assessment for cybersecurity of electric power systems. This project was part of the Young Scientist Summer Programme organized annually by IIASA.

Integrated safety assessment and justification of nuclear power plant automation (SAUNA)

OECD/NEA Working Group on Risk Assessment (WGRISK) activity “Comparative application of Digital I&C Modelling Approaches for PSA (DIGMAP)”, co-lead by Korea and Finland. (Tero Tyrväinen)

OECD/NEA WGHOF (Working Group on Human and Organisational Factors)

OECD/NEA WGHOF Task group on Achieving Reasonable Confidence in Validation Test Results of Integrated System Performance for Nuclear Power Plant Main Control Rooms

NUGENIA (Nugenia Generation II & III Association)

ISO/IEC JTC1 SC7 Software and systems engineering - WG7 Life cycle management; WG10 Process assessment.

OECD Halden Reactor Project, Man-Technology-Organization, Halden Programme Group, Janne Valkonen represents Finland

Collaboration with ITMO University (Russia) under the project "Development of methods, tools and technologies for design, verification and testing of reliable cyber-physical systems"

Safety of new reactor technologies (GENXFIN)

Generation IV International Forum (GIF)

IAEA coordinated research project on analysis of options and experimental examination of fuels for water-cooled reactors with increased accident tolerance (IAEA CRP ACTOF)

Electric Systems and Safety in Finnish NPP (ESSI)

Co-operation with Energiforsk GINO project, Sweden. Joint workshop with GINO project Steering group was organised at FORTUM, Keilaniemi, Espoo, 31.05.2018. Information exchange regarding the topic Open Phase Condition, Lightning overvoltages and Adaptive operation of NPP. Seppo Hänninen, Matti Lehtonen, Riku Pasonen and Antti Alahäivälä.

Comprehensive Analysis of Severe Accidents (CASA):

OECD/NEA BSAF-2 (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station, phase 2)

OECD/NEA THAI-3 (Thermal-hydraulics, Hydrogen, Aerosols, Iodine)

U.S.NRC CSARP (Co-operative Severe Accident Research Program)

NKS SPARC (Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics)

ASCOM (ASTEC COMmunity)

Chemistry and transport of fission products (CATFIS):

OECD/NEA STEM-2 (Source Term Evaluation and Mitigation)

OECD/NEA BIP-3 (Behaviour of Iodine)

NUGENIA TA2.4 Source term area

NUGENIA TA2.4 Integration of Pool scrubbing Research to Enhance Source-term Calculations (IPRESCA) project

Comprehensive and systematic validation of independent safety analysis tools (COVA):

OECD/WGAMA (Working Group on Analysis and Management of Accidents)

OECD/HYMERES-2 (Hydrogen Mitigation Experiments for Reactor Safety)

OECD/PKL-4 (Primary Coolant Loop Test Facility)

FONESYS (The Forum & Network of System Thermal-Hydraulic Codes in Nuclear Reactor Thermal-Hydraulics)

USNRC/CAMP (Code Applications and Maintenance Program)

Integral and separate effects tests on thermal-hydraulic problems in reactors (INTEGRA):

OECD/NEA PKL Phase 4 project

Development and validation of CFD methods for nuclear reactor safety assessment (NURESA):

OECD/NEA HYMERES

Nordic Thermal Hydraulic Network (Northnet), Roadmap 1 (OpenFOAM CFD-solver for nuclear reactor safety assessment).

OpenFOAM multiphase CFD solver development in co-operation with Helmholtz-Zentrum Dresden-Rossendorf and the OpenFOAM Foundation.

Nuclear criticality and safety analyses preparedness at VTT (KATVE):

OECD/NEA NSC (Nuclear Science Committee)

OECD/NEA WPNCS (Working Party on Nuclear Criticality Safety)

EWGRD (European Working Group on Reactor Dosimetry)

International Serpent users’ community (more than 800 users in 210 organizations around the world)

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

Collaboration with International Serpent user community (900 users in 217 universities and research organizations in 42 countries worldwide).

Participation in the activities of American Nuclear Society (ANS) Reactor Physics Division (RPD).

Membership in the Editorial Board of Annals of Nuclear Energy.

Membership in the Technical Program Committee of the PHYSOR 2018 international conference.

Membership in the Technical Program Committee of the M&C 2019 international conference.

Membership in the International Advisory Board of the SNA + MC2020 International Conference.

Membership in the OECD Nuclear Energy Agency, Working Party on Nuclear Criticality Safety (WPNCS), Expert Group on Advanced Monte Carlo Techniques (EGAMCT).

Membership in the OECD Nuclear Energy Agency, Working Party on Scientific Issues of Reactor Systems (WPRS).

Membership in the OECD Nuclear Energy Agency, Management Board on the Development, Application and Validation of Nuclear Data and Codes (MBDAV).

Membership in European Nuclear Society (ENS) High Scientific Council.

Physics and Chemistry of Nuclear Fuel (PANCHO)

OECD Halden Reactor Project, Fuel & Materials, Halden Programme Group, Ville Tulkki represents Finland

OECD/NEA Working Group on Fuel Safety (WGFS), Henri Loukusa

OECD/NEA WGFS RIA Fuel Code Benchmark Phase III, Asko Arkoma

OECD/NEA WGFS RIA State-of-the-art Report Update, Asko Arkoma

OECD/NEA CABRI International Project, Henri Loukusa

Uncertainty and sensitivity analyses for reactor safety (USVA):

Participation to the OECD Nuclear Energy Agency (NEA) Benchmark for Uncertainty Analysis in Modelling (UAM) for the Design, Operation and Safety Analysis of LWRs

Experimental and numerical methods for external event assessment improving safety (ERNEST)

Participation in ERNCIP European Reference Netweor for Critical Infrastructure Protection, Thematic group Resistacne of Structures to Explosition Effects

FIre Risk Evaluation and Defence-in-Depth (FIRED)

OECD/NEA PRISME3

Analysis of Fatigue and Other cUmulative ageing to exteND lifetime (FOUND)

NUGENIA Association Technical Area 8 (TA8), ENIQ (European Network for Inspection and Qualification) Task Group Risk (TGR) activities.

Co-operation with the Swedish-Finnish Beräkningsgrupp (BG)

Nugenia: Project ATLAS+ (Advanced structural integrity assessment tools for safe long term operation)

ASME PVP: Informal networking with the main contributors in the field of environmental fatigue.

Long term operation aspects of structural integrity (LOST):

ASTM E08 fatigue and fracture conference, Additive manufacturing

ECF22, European Conference on Fratcure,

PVP2018, Pressure vessel and piping conference

Mitigation of cracking through advanced water chemistry (MOCCA):

European Co-operation Group on Corrosion Monitoring (ECG-COMON)

International Co-operative Group on Environmentally Assisted Cracking (ICG-EAC)

The International Association for the Properties of Water and Steam (IAPWS)

Thermal ageing and EAC research for plant life management (THELMA):

International Co-operative group on environmentally assisted cracking, ICG-EAC (U. Ehrnstén).

Round Robin on initiation in Alloy 600 arranged by the ICG-EAC group (A. Toivonen)

EU H2020 project INCEFA+ - Increasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment (C. Huotilainen).

Co-operation with doctoral student Martin Bjurman, KTH and professor P. Efsing on thermally aged cast stainless steel (U. Ehrnstén).

Co-operation with professor Young Suk Kim, KAERI, Korea on thermally aged Alloy 690 (R. Mouginot).

Co-operation with professor M. Short, MIT, USA on thermally aged weld metals (H. Hänninen).

Nugenia TA4 - Integrity assessment & ageing of systems, structures & components (U. Ehrnstén)

OECD Halden project (M. Ivanchenko)

NDE of NPP primary circuit components and concrete infrastructure (WANDA)

International cooperation within U.S. Nuclear Regulatory Commission (NRC) PIONIC (Program for Investigation of NDE by International Collaboration)

Ferreira, M., ODOBA Project collaboration, 18.4.2018, Paris, France

Ferreira, M & Al-Neshawy, F., NDE Seminar - Wanda-Energiforsk Whorkshop, 4.9.2018, Otaniemi, Finland

Ferreira, M., ODOBA Project collaboration, 5-6.11.2018, Aix-en-Provence, France

Ferreira, M., ICIC collaboration, 12-14.11.2018, Knoxville, USA

A special guest lecture on 25.10.2018 by Research Professor, Dr. rer. nat. Ernst Niederleithinger, the Head of Division 8.2 (Non-Destructive Damage Analysis and Environmental Measurement Methods), BAM, Berlin, Germany. The lecture topic: Non-destructive evaluation for Nuclear Power Plant concrete infrastructure.

Condition Monitoring, Thermal and Radiation Degradation of Polymers Inside NPP Containments (COMRADE):

3rd annual COMRADE workshop, 4 – 5th December 2017, Forsmark, Sweden.

Participation in Horizon 2020 project "TeaM CABLES - European Tools and Methodologies for an efficient ageing management of nuclear power plant cables".

Evolving the Fennoscandian GMPEs (EVOGY)

OECD/NEA, Working Group on Integrity of Components and Structures (IAGE), Seismic Behavior of Components and Structures sub-group. Particpation as member.

OECD/NEA, Leading the CAPS project: "*Comparison of PSHAs in areas with different level of seismic activity (CompPSHA)*", Final report accepted by the CSNI group in Dec. 2018 – In editorial stage with OECD

NKS activity "*Synthetic ground motions to support the Fennoscandian GMPEs (SYNTAGMA)*", embeded with EVOGY

JHR collaboration & Melodie follow-up (JHR):

Jules Horowitz Reactor Working Groups

Barsebäck RPV material used for true evaluation of embrittlement (BRUTE)

International co-operative group on radiation embrittlement, IGRDM

BREDA – Barsebäck Research and Development Arena.

Co-operation with Kungliga Tekniska Högskolan, KTH, Stockholm, Sweden, with professor Pål Efsing and Doctoral Student Magnus Boåsen

Co-operation with Chalmers University, Gothenburg, Sweden, with professor Mattias Thuvander and Doctoral Student – Doctoral student, now postdoc researcher Kristina Lindqvist.

Appendix 3

Academic degrees obtained in the projects in 2018

Extreme weather and nuclear power plants (EXWE)

Master of Science in Philosophy:

Korpinen, A. Rajakerroksen simulointi heterogeenisellä alustalla. Master’s thesis, University of Helsinki, Institute of Atmospheric and Earth System Research, December 2018.

Electric Systems and Safety in Finnish NPP (ESSI)

Master of Science in Technology:

Ismet Tuna Gürbüz: Lightning Induced Over-voltages in Nuclear Power Plants. Master´s thesis. Aalto School of Electrical Engineering, December 2018.

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

Master of Science in Technology:

Ilkka Tahvola, Modelling of PKL test facility with TRACE code, Lappeenranta University of Technology, 2018

Physics and Chemistry of Nuclear Fuel (PANCHO)

Doctor of Technology:

Asko Arkoma: Modelling design basis accidents LOCA and RIA from the perspective of single fuel rods, Aalto University, Department of Applied Physics, date of defence: 8.3.2018

Long term operation aspects of structural integrity (LOST)

Master of Science in Philosophy:

Pentti Arffman: Mechanical properties of dissimilar metal welds, Helsinki University, April 2019

Master of Science in Technology:

Sirkiä Laura: Laura Sirkiä, Applicability of miniature Compact Tension specimens for fracture toughness determination in ductile-brittle transition range, Master’s Thesis, Espoo, Aalto University, Department of Engineering Materials, date of the graduation 30.11.2017

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

Master of Science in Technology:

Oskari Jessen-Juhler: Artificial Flaw Simulation in Ultrasonic Inspection of Austenitic Stainless Steel Weld, September 2018

Evolving the Fennoscandian GMPEs (EVOGY)

Bachelor of Sience in Philosophy:

Riina Aapasuo, BSc thesis on the EVOGY database of Fennoscandian earthquake recordings, “Laitevastuksen poisto seismologisessa signaalinkäsittelyssä”

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Bachelor of Science in Technology:

Siiri Jämsen: Void fraction measurement with wire-mesh sensor in swirling two-phase flow, Lappeenranta University of Technology, School of Energy Systems, March 2018

Appendix 4

International travels in the projects in 2018

Crafting operational resilience in nuclear domain (CORE)

Laarni, J., NUCLEAR DAYS & NUGENIA Forum 2018, Prague, Czech Republic, 10-11 April, 2018.

Laarni, J., 12th Multi Conference on Computer Science and Information Systems (MCCSIS 2018), Madrid, Spain, 17 – 20 July, 2018.

Teperi, A.-M., 32nd Triennal Congress of the International Commission on Occupational Health (ICOH), Dublin, Ireland, 29 April – 4 May, 2018.

Teperi, A.-M., 9th AHFE International Conference, Florida, Orlando, USA, 21–25 July, 2018.

Extreme weather and nuclear power plants (EXWE)

Jylhä, K. American Meteorological Society (AMS) Annual Meeting 2018, Austin, USA, 7–11 January, 2018

Lehtonen, I., European Meteorological Society (EMS) Annual Meeting: European Conference for Applied Meteorology and Climatology 2018, Budapest, Hungary, 3-7 September 2018.

Leijala, U., Nilsen, J.E.Ø., Ravndal, O., Sande, H., and Chafik, L.: Investigating connection between decadal mean sea level variations and extreme sea levels on the Norwegian west coast. Poster presentation at the American Geophysical Union Fall Meeting 2018, Washington D.C., USA, 10–14 December 2018.

Leijala, U., Nilsen, J.E.Ø., Ravndal, O., and Sande, H.: The effect of decadal variability on the extreme sea levels on the west coast of Norway. Poster presentation at the European Geosciences Union General Assembly 2018, Vienna, Austria, 8–13 April 2018.

Leijala, U., Björkqvist, J.-V., Johansson, M.M., Pellikka, H., Laakso, L., Kahma, K.K.: Quantifying return periods of rare coastal floods by studying the joint effect of mean sea level change, short-term sea level variations and wind waves. Poster presentation at the American Geophysical Union Ocean Sciences Meeting 2018, Portland, Oregon, USA, 11–16 February 2018.

Luomaranta, A., European Meteorological Society (EMS) Annual Meeting: European Conference for Applied Meteorology and Climatology 2018, Budapest, Hungary, 3-7 September 2018.

Pellikka, H., European Geosciences Union (EGU) General Assembly 2018, Vienna, Austria, 8–13 April 2018

Toivonen E., European Geosciences Union (EGU) General Assembly 2018, Vienna, Austria, 8–13 April 2018.

Management Principles and Safety Culture in Complex Projects (MAPS)

Gotcheva, N. Invited participation at IAEA meeting to finalize the Safety Culture Continuous Improvement Process (SCCIP) Training Material, Vienna, Austria, 9-11 April 2018.

Kujala, J. and Artto, K. participated in 9th International Project Business Workshop University of Leeds, Leeds, UK, 1-2 June 2018.

Viitanen, K. participated in international scientific conference Probabilistic Safety Assessment & Management conference (PSAM14), September 16-21, 2018, CA, Los Angeles, USA.

Kujala, J., Aaltonen, K. and Pargar, F. participated in the International Research Network on Organizing by Projects (IRNOP) conference, RMIT University, Melbourne, Australia, 10-12 December 2018.

Probabilistic risk assessment method development and applications (PRAMEA)

Jan-Erik Holmberg. IAEA Consultancy Meeting on the development of the Safety Report on Human Reliability Assessment for Nuclear Installations, Vienna, 19-22 February, 2018.

Jan-Erik Holmberg. SITRON project meetings, Stockholm, 16–17 April, 2018.

Jan-Erik Holmberg. OECD international workshop on status of site level PSA (including multi-unit PSA) developments, Munich, Germany, July 18–20, 2018.

Jan-Erik Holmberg. SITRON project meetings, Stockholm, 21–22 August, 2018.

Jan-Erik Holmberg. Nordic Nuclear and Radiation Risk Estimates — Advances and Uncertainties — Joint NKS-R and NKS-B Seminar, Finlandshuset, Stockholm, 15–16 January 2019.

Jan-Erik Holmberg and Tero Tyrväinen. NKS-R SITRON: Workshop on site risk analysis for nuclear power plants, Stockholm, 17 January, 2019.

Markus Porthin. OECD/NEA Working Group on Risk Assessment (WGRISK) 19th Annual Meeting, Paris, France, 7 - 9 March 2018.

Alessandro Mancuso, Young Scientist Summer Programme, International Institute of Applied Systems Analysis, Laxemburg, Austria, 1 June – 31 August 2018.

Alessandro Mancuso, European Conference on Operational Research, EURO 2018, Valencia, 8-11 July 2018.

Integrated safety assessment and justification of nuclear power plant automation (SAUNA)

Porthin, M., 19th Annual Meeting of the CSNI Working Group on Risk Assessment (WGRISK), March 7–9, 2018, Paris, France.

Vyatkin, V., Workshop on formal methods at ITMO University, May 2–5, Saint-Petersburg, Russia.

Valkonen, J., OECD Halden Reactor Project, Man-Technology-Organization, Halden Programme Group meeting, May 15–16, 2018, Zürich, Switzerland.

Vyatkin, V., 27th IEEE International Symposium on Industrial Electronics (ISIE 2018), June 13–15, 2018, Cairns, Australia.

Holmberg, J.-E., European Safety and Reliability Conference ESREL 2018, June 17–21, 2018, Trondheim, Norway.

Pakonen, A., Papakonstantinou, N., Vyatkin, V., 16th IEEE International Conference on Industrial Informatics (INDIN 2018), July 18–20, 2018, Porto, Portugal.

Papakonstantinou, N., ASME 2018 International Design Engineering Technical Conferences & Computers and Information in Engineering Conference (IDETC/CIE 2018), August 26–29, 2018, Quebec City, Canada.

Varkoi, T., 25th European Conference on Systems, Software and Services Process Improvement (EuroSPI 2018), September 5–7, 2018, Bilbao, Spain.

Vyatkin, V., Workshop on bounded model checking at ITMO University, September 27–30, Saint-Petersburg, Russia.

Vyatkin, V., Seminar on user-friendly interpretation of model-checking results, ITMO University, September 27–30, Saint-Petersburg, Russia.

Vyatkin, V., Annual Conference of IEEE Industrial Electronics Society, October 19–26, Washington, USA.

Vyatkin, V., Congress and exhibition SPS/IPC/Drives, November 26–27, Nuremberg, Germany.

Tyrväinen, T., CSNI Working Group on Risk Assessment (WGRISK), DIGMAP task meeting, January 28–30, 2019, Munich, Germany.

Papakonstantinou, N., Reliability and Maintainability Symposium (RAMS 2019), January 28–31, 2019, Orlando, FL, USA.

Safety of new reactor technologies (GENXFIN)

Penttilä, S., 12th Information exchange meeting on supercritical water cooled reactors R&D, March 19–20, 2018, Montreal, Canada.

Tulkki, V., IAEA technical meeting on the deployment of non-electric applications using nuclear energy for climate change mitigation, April 16–18, 2018, Vienna, Austria.

Tulkki, V., American Nuclear Society annual meeting, June 17–21, 2018, Philadelphia, Pennsylvania, USA.

Penttilä, S., Generation IV International Forum supercritical water cooled reactors materials & chemistry meeting, September 6–7, 2018, Madrid, Spain.

Penttilä, S., International conference on generation IV and small reactors, November 6–8, 2018, Ottawa, Canada.

Penttilä, S., IAEA coordinated research project meeting on accident tolerant fuels, November 26–30, 2018, Vienna, Austria.

Electric Systems and Safety in Finnish NPP (ESSI)

Lehtonen, M. 19th International Scientific Conference on Electric Power Engineering, EPE 2018, 16-18 May 2018, Brno, Czech Republic.

Hänninen, S. Energiforsk Annual Nuclear Conference 2019 - Flexible nuclear power and ancillary services, 23 - 24 Jan 2019, Stockholm, Sweden.

Practical applications and further development of Overall Safety Concept (ORSAPP)

Society for Risk analysis, Australia -New Zealand) SRA-ANZ 2018 Annual conference, 26th-27th of September 2018, Sydney, Australia.
Oral presentation: Decommissioning of nuclear installations as a challenge to risk governance - Call for a sociotechnical perspective.

Comprehensive Analysis of Severe Accidents (CASA)

Sevón, Tuomo. OECD BSAF-2 meeting. 22–26 January 2018. Paris, France.

Sevón, Tuomo. CSARP/MCAP meeting. 5-8 July 2018. Rockville, USA.

Nieminen, Anna & Strandberg, Magnus. THAI-3 PRG5 and MB5 meetings. 1-2 October 2018. Frankfurt, Germany.

Nieminen, Anna. IPRESCA in-kind project meeting. 3 October 2018. Frankfurt, Germany.

Nieminen, Anna. ASCOM in-kind project kick-off meeting. 8-10 October 2018. Aix-en-Provence, France.

Chemistry and transport of fission products (CATFIS)

Kärkelä, T., Gouëllo, M., OECD/NEA STEM-2 meeting (Programme Review Group and Management Board), Paris, France, June 2018.

Gouëllo, M., Kärkelä, T., OECD/NEA BIP-3 meeting (Programme Review Group and Management Board), Paris, France, June 2018.

Kärkelä, T., Integration of Pool scrubbing Research to Enhance Source-term Calculations (IPRESCA) second meeting, Frankfurt, Germany, October 2018.

Gouëllo, M., International Congress on Advances in Nuclear Power Plants (ICAPP 2018), Charlotte, USA, April 8-11, 2018.

Gouëllo, M., ASTEC training at IRSN for three weeks, France, November 2018.

Comprehensive and systematic validation of independent safety analysis tools (COVA)

Hillberg S. U.S. NRC CAMP meeting, 10.-14.12.2018 , Bethesda, USA.

Karppinen, I. OECD/NEA HYMERES-2 meeting, 22.-24.3.2018, Villigen, Switzerland

Karppinen, I. OECD/NEA PKL-4 meeting, 23.-24.5.2018, Lappeenranta, Finland

Karppinen, I. OECD/NEA WGAMA meeting, 17.-20.9.2018, Paris, France

Szogradi, M. OECD/NEA PKL-4 workshop & PRG meeting, 5.-9.11.2018, Barcelona, Spain

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

Heikki Purhonen, Vesa Riikonen, The Programme Review Group and Management Board meetings of the OECD/NEA PKL Phase 4 Project, Barcelona, Spain, 6th November 2018.

Nuclear criticality and safety analyses preparedness at VTT (KATVE)

Kotiluoto, P., OECD NEA Nuclear Science Committee meeting, 12-14th June, NEA Headquarters, Boulogne-Billancourt, France

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

April 22-26, 2018, Cancun, Mexico – PHYSOR 2018 and Serpent workshop (Serpent developer team, Sahberg, Rintala)

May 29 - June 1, 2018, Espoo, Finland – 8th International Serpent User Group meeting

June 17-21, 2018, Philadelphia, PA, USA – ANS Annual Meeting and Reactor Physics Division Executive Committee Meeting (Leppänen)

November 11-15, 2018, Orlando, FL, – ANS Winter Meeting and Reactor Physics Division Executive Committee Meeting (Leppänen)

Physics and Chemistry of Nuclear Fuel (PANCHO)

Arkoma, A. OECD/NEA RIA Benchmark Phase III kick-off meeting and RIA State-of-the-art Report Update meeting, March 12-13, 2018, Paris, France.

Loukusa, H. OECD/NEA WGFS Interim Meeting, March 15, 2018, OECD Boulogne, Paris, France.

Tulkki, V. 158th Halden Programme Group Meeting, May 15-16, 2018, Zurzach, Switzerland.

Tulkki, V. Extraordinary Halden Programme Group Meeting, August 23-24, 2018, Halden, Norway.

Loukusa, H. OECD/NEA WGFS Plenary Meeting and Technical Opinion Paper on Accident Tolerant Fuels, kick-off meeting, September 18-21, 2018, OECD Boulogne, Paris, France.

Arkoma, A., OECD/NEA WGFS RIA code benchmark phase 3, 2nd meeting, Update of the RIA state-of-the-art report (SOAR), September 17-19, 2018, OECD Boulogne, Paris, France.

Loukusa, H., TopFuel 2018 conference and FRAPCON/FRAPTRAN User's Group meeting, September 30 - October 4, 2018, Prague, Czechia.

Tulkki, V., Building a Multinational Fuel and Materials Testing Capacities for Science, Safety and Industry, OECD/NEA workshop, October 4-5, Paris, France.

Heikinheimo, J., NuMat 2018 conference, October 14-18, Seattle, WA, USA.

Heikinheimo, J., 159th Halden Programme Group Meeting 2018, October 23-24, 2018, Halden, Norway.

Safety analyses for dynamical events (SADE)

Sahlberg, V., AER working group D on VVER reactor dynamics and safety, June 12–13, 2018, Dresden, Germany

Syrjälahti, E., Kick-off meeting of OECD/NEA Rostov-2 benchmark, 18 May, 2018, Lucca, Italy

Uncertainty and sensitivity analyses for reactor safety (USVA)

Arkoma, A., ANS Best Estimate Plus Uncertainty International Conference (BEPU2018), May 13-18, 2018, Lucca, Italy.

Valtavirta, V., ANS Best Estimate Plus Uncertainty International Conference (BEPU2018), May 13-18, 2018, Lucca, Italy.

Syrjälahti, E., 12th workshop of Light Water Reactor Uncertainty Analysis in Modelling Benchmark (UAM-LWR), May 16-17, 2018, Lucca, Italy.

FIre Risk Evaluation and Defence-in-Depth (FIRED)

Sikanen Topi, Hostikka Simo: Participation to OECD/NEA PRISME meeting 18-20.4.2018 in Aix-en-Provence, France.

Vaari Jukka: Participation to 22nd International Symposium an Analytical and Applied Pyrolysis (Pyro2018), Kyoto, Japan, June 3-8, 2018

Matala Anna, Hostikka Simo: Participation to OECD/NEA PRISME meeting 11-14.11.2018 in Aix-en-Provence, France.

Analysis of fatigue and other cumulative ageing to extend lifetime (FOUND)

Cronvall, O. ASME PVP Conference July 15–20, 2018 Prague, Czech Republic

Cronvall, O. Kärnteknik 2018 Conference, November 26-28, 2018. Solna, Sweden.

Kuutti, J. ASME PVP Conference July 15–20, 2018 Prague, Czech Republic

Oinonen, A. NUGENIA TA8: ENIQ - Sub-Area Risk (SAR) meeting, 13.-15.3.2018, Erlangen, Deutschland.

Seppänen, T. ASME PVP Conference July 15–20, 2018 Prague, Czech Republic

Timperi, A. ASME PVP Conference July 15–20, 2018 Prague, Czech Republic

Long term operation aspects of structural integrity (LOST)

Lindqvist, S. ASTM E08 fatigue and fracture conference, 2-6th May, San Diego, USA.

Lindqvist, S. ASTM E08 fatigue and fracture conference, 14-18th November, Washington D.C, USA.

Lindqvist, S., ECF22, 22nd European Conference on Fracture, 20-24th June, Serbia, Belgrade.

Mitigation of cracking through advanced water chemistry (MOCCA)

Saario, T., project meeting, June 25-28, 2018, Sofia, Bulgaria.

Sipilä, K., 17th Nordic Corrosion Congress, May 23-25, 2018, Copenhagen, Denmark.

Sipilä, K., European Co-operative Group on Corrosion Monitoring (ECG-COMON) meeting, June 11-12, 2018, Ljubljana, Slovenia.

Sipilä, K., Nuclear Plant Chemistry (NPC 2018) conference, September 9-14, 2018, San Francisco, USA

Thermal ageing and EAC research for plant life management (THELMA)

Ehrnsten, Ulla & Huotilainen, Caitlin. International Co-operative Group on Environmentally Assisted Cracking, ICG-EAC 2018, Steering committee meetings, presentation of 2020 meeting in Finland and yearly meeting. Knoxville, US, 15-20.04.2018.

Huotilainen, Caitlin. 3rd International Conference on Metals&Hydrogen, Ghent, Belgium, 29-31.05.2018

Ivenchenko, Mykola. Contribution of Materials Investigations and Operating Experience to Light Water NPPs' Safety, Performance and Reliability, Fontevraud 8, Avignon, France, 17-20.9. 2018.

Ahonen, Matias. EPRI 690/52 expert group meeting. Tampa, US, 28-30.11.2018.

NDE of NPP primary circuit components and concrete infrastructure (WANDA)

Leskelä, E., Program for Investigation of NDE by International Collaboration (PIONIC) meeting 2, April 23-26, 2018, Pacific Northwest National Laboratory (PNNL), Richland, WA 99352, USA.

Koskinen, T., 12th European Conference of Non-destructive Testing (12th ECNDT), June 11-15, 2018, Gothenburg, Sweden.

Al-Neshawy, F., 12th European Conference of Non-destructive Testing (12th ECNDT), June 11-15, 2018, Gothenburg, Sweden.

Fahim Al-Neshawy – The NDT&E Advanced Training Workshop, June 27-July 03, 2018. BAM, Berlin, Germany.

Virkkunen, I., Program for Investigation of NDE by International Collaboration (PIONIC) meeting 3, November 5-8, 2018, Mitsubishi Heavy Industries, Osaka, Japan

Ferreira, M., ODOBA Project collaboration, 18.4.2018, Paris, France

Ferreira, M., ODOBA Project collaboration, 5-6.11.2018, Aix-en-Provence, France

Ferreira, M., ICIC collaboration, 12-14.11.2018, Knoxville, USA

Condition Monitoring, Thermal and Radiation Degradation of Polymers Inside NPP Containments (COMRADE)

Bondeson, A., Jansson, A., Sipilä, K., Vaari, J. COMRADE workshop, 4-5th December 2018, Forsmark NPP, Forsmark, Sweden.

Jansson, A. FONTEVRAUD 9 17-20 September 2018, Avignon, France.

Jansson A, Molander, Workshop on NPP, Ringhals, Värö, 1st March 2018

Jansson A, Molander M Workshop NPP, Forsmark 21st March 2018

Molander M, Lindhagen E, Workshop in NPPs, OKG, Oskarshamnn, 4th April 2018

Evolving the Fennoscandian GMPEs (EVOGY)

L. Fülöp, annual meeting of the Working Group on Integrity and Ageing of Components and Structures (WGIAGE), seismic engineering sub-group.

V. Jussila, Synthetic ground motions to support the near-field seismic hazard prediction in Fennoscandia, Stockholm NKS seminar 15-16 January 2019

P. Mäntyniemi, T. Vuorinen: 49^{th} Nordic Seismology Seminar, Lillestrøm, Norway 24-26 September 2018

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Hujala Elina, Pyy Lauri, Short Courses on Multiphase Flows, February 12-16, 2018, ETH Zurich, Switzerland.

JHR collaboration & Melodie follow-up (JHR)

V. Tulkki, OECD/NEA workshop on materials testing infrastructure, 10.1.2018, Paris, France.
V. Tulkki, C. Huotilainen, P. Kinnunen, 8^{th} Technical seminar, 21.-22.3.2018 Cadarache, France.
V. Tulkki, OECD/NEA 2^{nd} workshop on materials testing infrastructure, 4.-5.10.2018, Paris, France.
V. Tulkki, C. Huotilainen, P. Kinnunen, JHR WG meeting, 9.-10.10.2018, Oxford, UK.

Radiological laboratory commissioning (RADLAB)

Lavonen, T. ja Myllykylä, E., Nordic Conference on Plasma Spectrochemistry, 9-14.6.2018, Loen, Norja

Jokipii, M. ja Tähtinen, S., Factory acceptance test of radioactive specimen storage at ITD, 23-24.8.2018, Dresden, Saksa

Heikola T., ICP-MS operator training, 3-7.9.2018, Bremen, Saksa

Barsebäck RPV material used for true evaluation of embrittlement (BRUTE)

Lydman, J. Visit to Manchester University, England, to learn Focussed Ion Beam technique 9-13.7.2018.

Ehrnstén, U. NKS Seminar Nordic Nuclear and Radiation Risk Estimates - Advances and Uncertainties, Stockholm, Sweden, 16 January 2019.