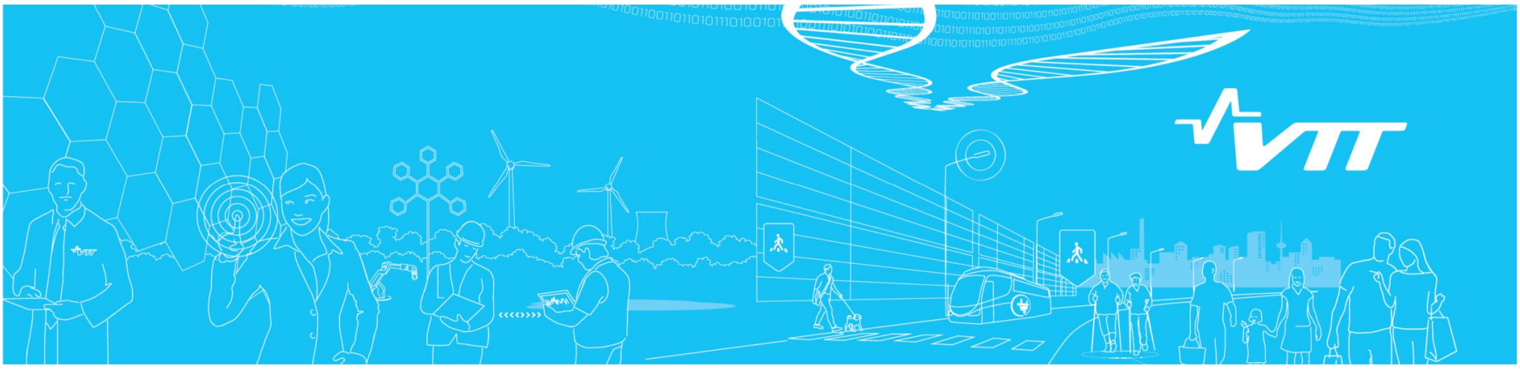
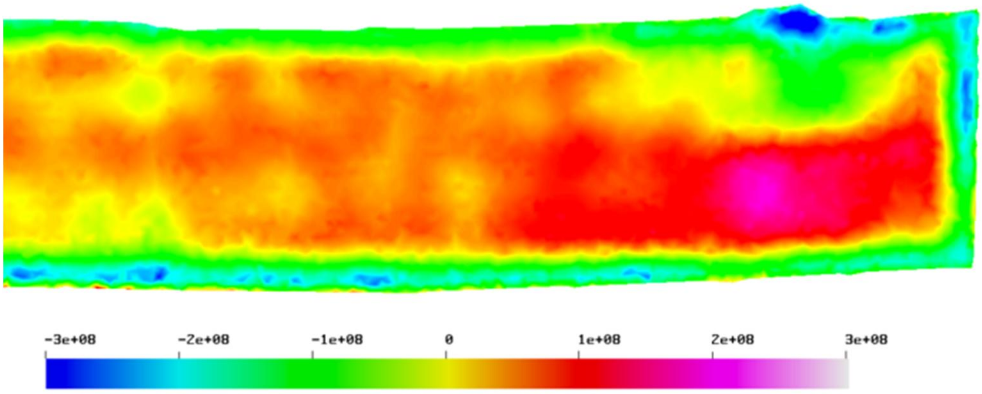


RESEARCH REPORT

VTT-R-02893-17



SAFIR2018 Annual Report 2016

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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), Finnish Funding Agency for Innovation (Tekes), and Swedish Radiation Safety Authority (SSM).

The actual volume of the SAFIR2018 programme in 2016 was 7,0 M€ and 50 person years. Main funding organisations in 2016 were the Finnish State Waste Management Fund (VYR) with 4,1 M€ and VTT with 1,5 M€. The programme was divided into three research areas and in 2016 research was carried out in 28 projects.

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

Confidentiality	Public	
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Appendices

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 be available for publication and their exploitation shall not be restricted to the power plants of a single licence holder.

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:

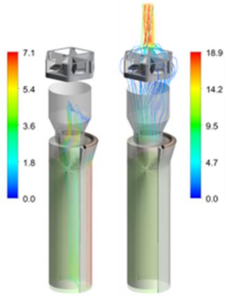
> *The SAFIR2018 research community is a vigilant, internationally recognised and strongly networked competence pool that carries out research on topics relevant to the safety of Finnish nuclear power plants on a high scientific level and with modern methods and experimental facilities.*

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

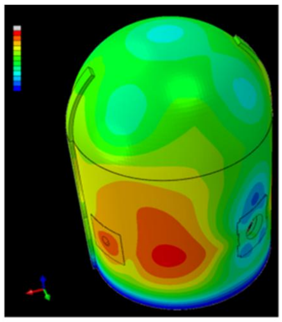
SAFIR2018 management board was nominated in September 2014. It consists of representatives of the Radiation and Nuclear Safety Authority (STUK), the Ministry of Economic Affairs and Employment (MEAE), Fennovoima Oy, Fortum, Teollisuuden Voima Oyj (TVO), Technical Research Centre of Finland Ltd (VTT), 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 public call for research proposals for 2016 was announced in the beginning of September 2015. 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 2016. The funding decisions were made by the Finnish State Nuclear Waste Management Fund (VYR) in March 2016. In 2016 the programme consisted of 28 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 2016 were 6,8 M€ and 7,0 M€ and 44 and 50 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, and Appendix 5 contains list of the persons involved in the programme in the Management Board, Steering Groups, Reference Groups and in the projects.

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 2016

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 2016, the research was performed in altogether 28 research projects. The total volume of the programme was 7,0 M€ and 50 person years. The research projects in the various research areas with their planned and realised volumes are given in Table 2.1.

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

Table 2.1 SAFIR2018 projects in 2016.

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	249,0	242,7	21	21
	Extreme weather and nuclear power plants	EXWE	FMI	244,7	327,2	23	40
	Management principles and safety culture in complex projects	MAPS	VTT, Aalto, University of Oulu, University of Jyväskylä	204,0	203,2	16	17
	Probabilistic risk assessment method development and applications	PRAMEA	VTT, Aalto, Risk Pilot	343,0	342,1	27	28
	Integrated safety assessment and justification of nuclear power plant automation	SAUNA	VTT, Aalto, FISMA, Risk Pilot, IntoWorks	455,0	472,6	36	42
	Safety of new reactor technologies	GENXFIN	VTT	112,0	112,0	6	7
2. Reactor safety							
	Comprehensive analysis of severe accidents	CASA	VTT	228,0	227,9	15	16
	Chemistry and transport of fission products	CATFIS	VTT	158,0	157,8	9	10
	Comprehensive and systematic validation of independent safety analysis tools	COVA	VTT	273,0	278,4	18	19
	Couplings and instabilities in reactor systems	INSTAB	LUT	157,0	167,0	13	17
	Integral and separate effects tests on thermal-hydraulic problems in reactors	INTEGRA	LUT	359,0	359,9	29	30
	Nuclear criticality and safety analyses preparedness at VTT	KATVE	VTT	178,5	178,4	14	14
	Development of a Monte Carlo based calculation sequence for reactor core safety analyses	MONSOON	VTT	126,0	126,3	10	11
	Development and validation of CFD methods for nuclear reactor safety assessment	NURESA	VTT, Aalto, LUT	236,0	236,3	17	16

	Physics and chemistry of nuclear fuel	PANCHO	VTT	277,0	277,1	20	22
	Safety analyses for dynamical events	SADE	VTT	100,0	100,2	8	9
	Uncertainty and sensitivity analyses for reactor safety	USVA	VTT, Aalto	140,0	140,0	13	13
3. Structural safety and materials							
	Experimental and numerical methods for external event assessment improving safety	ERNEST	VTT	89,0	89,5	5	5
	Fire risk evaluation and Defence-in-Depth	FIRED	VTT, Aalto	195,0	195,3	18	16
	Analysis of fatigue and other cumulative ageing to extend lifetime	FOUND	VTT, Aalto	365,0	409,2	28	29
	Long term operation aspects of structural integrity	LOST	VTT	224,5	244,6	14	16
	Mitigation of cracking through advanced water chemistry	MOCCA	VTT	147,0	148,1	10	9
	Thermal ageing and EAC research for plant life management	THELMA	VTT, Aalto	268,0	276,8	19	19
	Non-destructive examination of NPP primary circuit components and concrete infrastructure	WANDA	VTT, Aalto	91,3	91,3	6	6
	Condition monitoring, thermal and radiation degradation of polymers inside NPP containments	COMRADE	VTT, SP	191,0	193,8	9	9
4. Research infrastructure							
	Development of thermal-hydraulic infrastructure at LUT	INFRAL	LUT	308,0	313,5	16	20
	JHR collaboration & Melodie follow-up	JHR	VTT	29,0	30,5	2	2
	Radiological laboratory commissioning 2016	RADLAB	VTT	750,0	750,5	34	47
0. Programme administration							
	SAFIR2018 administratiion and information	ADMIRE	VTT	330,0	330,0	11	11

The costs of ADMIRE are for period 1.1.2016-31.3.2017. The costs include two subcontracted small study projects and value-added tax 24%.

2.1 Plant safety and systems engineering

In 2016 the research area “Plant safety and systems engineering” consisted of six 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).

2.1.1 CORE - Crafting operational resilience in nuclear domain

The aim of the CORE project 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 2016

First, the literature review (prepared in 2015) was used to design empirical studies and served as a starting point for developing the principles for identifying successful actions and decisions in NPP context. A preliminary framework for learning from successes was further elaborated in the form of a guideline that proposes eight basic principles on how to learn from successes and describes a generic process for performing an analysis of successes. We arranged a workshop to discuss the practical potential of the principles for capturing successful events in operating experience work. The workshop also allowed us to further refine our findings and to align our conception of the methodology for capturing successful adaptations with the requirements of NPP practitioners.

The second goal was to create a method for the development of operator training. Based on the workshops held in 2015, we could identify several training needs and we could achieve understanding on what kind of training methods could be introduced to the operating organization. Based on these insights, we were able to make realistic suggestions of a new training method or modification to the existing training practices. The aim was to select and test the best training method options. Two methods for evaluation were applied: 1) workshop discussions with the training developers (for selecting and further developing a method option) 2) preliminary testing with the operators (one method option was tested here). Two workshops were conducted with the training developers, and the preliminary method was tested after four simulator training sessions of two different types, with four different training shifts. The preliminary results were promising: operators were positive about the selected training method, this being, a simulator self-evaluation method based on structured individual and group discussions.

Third, we have reviewed tools available for operator work modelling, and we have outlined a modelling tool for the analysis of collaborative troubleshooting within an operator crew and applied it to the analysis of a complex simulator scenario run. The approach describes the progress and evolution of a CR operator crew's knowledge states throughout the critical sections of a simulator run. We have presented its application to the analysis of a challenging simulated accident scenario in which only one operator crew of six crews was able to troubleshoot the problem successfully. We have also analysed operator practices in simulator conditions with the Functional Situation Modelling (FSM) method, and the FSM method has been further developed. We have also compared the FSM model data to operator actions that have happened in simulated accident and incident situations using time-stamped annotations in Noldus Observer software and considered this data in regards to operator stress (heart-rate variability, HRV) data.

Fourth, the first level data analysis on physiological and experienced stress as well as predicted load during the simulations was carried out, and the amount of physical activity (recorded with the motion sensors of the ECG measurement devices) during the recordings was analysed in order to control for the effects of operators moving within the simulator room. In addition, education of the operators on the effects of stress continued in the form of lectures at TVO Olkiluoto “Käytön koulutuspäivät” presenting the results gained from the simulations carried out on 2015. Emphasis was on presenting the measured stress levels during the simulations, evaluaton of crew performance and possible effects of stress on performance on the basis of literature. Overall, our efforts have been focused on integrating the information gained from the simulator study to existing knowledge on operator training, and on distribution and dissemination of the findings.

Fifth, the role of Technical Support Centre (TSC) of the two operating nuclear power plants in the emergency exercises were concentrated on. The research was conducted by interviewing the representatives of the two NPPs. The interviews focused on responsibilities, tools and practices of emergency exercises, regarding especially collaboration and coordination. In addition, the method for observing emergency exercises was developed.

Sixth, we assessed human contribution in the current safety management system in nuclear industry by analyzing interview data regarding currently used Human Factors (HF) procedures and by conducting safety documentation analysis regarding current HF guidelines and current practices (e.g. YVL guide, guidelines at each NPP). It was revealed that currently, conceptions regarding HF are individual and error based which may hinder the awareness and improvements at the organizational and system level of nuclear industry. We found that there is a need for more concrete and holistic HF-tools and models, representing macro-ergonomics and a new paradigm, positive safety thinking (Safety-II). We then modified and tested a HF tool as an investigation method in analyzing HF successes and weaknesses at different system levels at the background of operative events (OE). We accordingly analysed the interview data with user experiences as well as intervention material from two workshops, in which HF tool was tested. According to our results, the new HF tool offered a more accurate picture of the HFs affecting OEs, moving focus from individual errors to all layers of system safety as well as on successes and the factors that maintain safety. Users found the HF tool clear and easy to use and useful for OE investigation.

Deliverables in 2016

- A set of guidelines proposing eight basic principles on how to learn from successes and describing a generic process for performing an analysis of successes.
- A method for the development of operator training.
- A review of work-modelling tools and a modelling approach suitable for analysing collaborative diagnostic reasoning and troubleshooting of a NPP control room crew.
- Functional situational models of the TVO Olkiluoto simulation runs; specification of significant events in simulator runs and synching annotations to heart rate variability data on a timeline.

- Data analysis on physiological and experienced stress as well as predicted load during the simulations.
- Lectures/lecture notes on the topics of physiological stress, effects of stress on performance in general and specifically in critical situations and complex environments, as well as on stress management.
- A method for the observation of emergency exercises and for the analysis of the observation data.
- A refined version of the HF Tool and further evidence of its applicability to accident investigation in the nuclear domain.

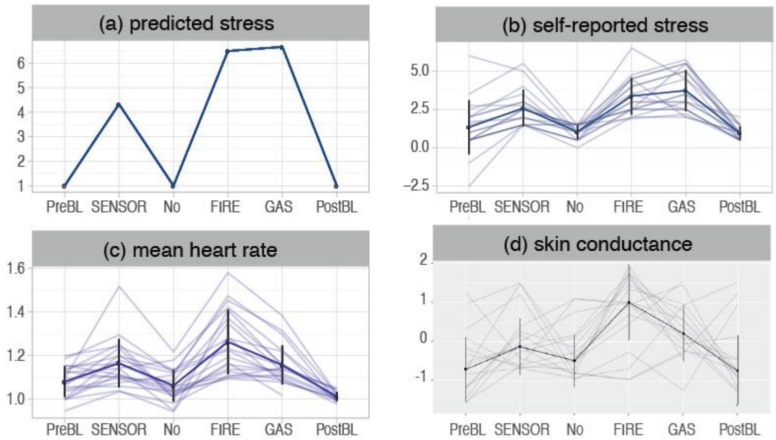


Figure 2.1.1.1. (a) Predicted stress (b) self-reported stress (c) mean heart rate and e) skin conductance during different parts of the simulator training: pre and post baselines (PreBL, PostBL), normal operation (No), malfunction of the oil pressure sensor (SENSOR), fire and radioactive steam leakage (FIRE, GAS).

2.1.2 EXWE - Extreme weather and nuclear power plants

The general objective of the EXWE project is to give better estimates of probabilities of extreme geophysical events that affect the design principles of nuclear power plants (NPPs) and may pose external threats to the plants. Four themes were covered in 2016: 1) extreme weather incidents and 2) extreme sea level events, 3) extreme space weather; and 4) atmospheric dispersion and dose assessment of accidental releases. The safety assessments of the NPPs require not only observation-based frequencies for the extreme events, but also estimates on the probabilities of such extremely rare events that have not occurred during the past 100 years of observations. Therefore both instrumental records and model simulations are utilized in the work. The end-users are the power companies designing and running NPPs as well as the nuclear safety authorities defining the safety regulations for NPP constructions and operations.

Specific goals in 2016

The specific topics in 2016 were as severe warm-season convective weather phenomena, such as hail; intense sea-effect snowfall; freezing rain; latest findings on strong winds; meteotsunamis and other short-period extreme-sea-level events; simultaneous occurrence of high sea level and high wind waves; extreme space weather i.e. solar storms and geomagnetic disturbances, and preparations needed in meteorological high-resolution modelling to be linked with a state-of-the art dispersion modelling system.

Extreme convective weather (ECW) in summer: In the warm season, ECW is characterized by thunderstorms producing heavy rain, large hail, intensive lightning, strong wind gusts (downbursts) and tornadoes. All of these phenomena can cause damages to infrastructure. Historical time series of ECW events show cross correlations between summertime 1) thunderstorm days, lightning and heavy rainfall, 2) heavy rainfall and tornadoes and 3) large hail and tornadoes. Atmospheric conditions favouring significant hail (diameter of 5 cm or larger) could be clustered into four distinct synoptic classes, two of which resemble the patterns observed in tornado situations in Finland. Every third hail day had at least one tornado report and two of the three hail days had daily rainfall of 30 mm or more.

Sea-effect snowfall: Sea-effect snowfall is formed in a convective situation when cold air flows over a warm ice-free water surface. Water area acts as a source of heat and moisture. This generates shallow convection that induces small and intensive convective precipitation which can drift to the coast as snowbands. In the record-breaking sea-effect snowfall case at Merikarvia on 8 January 2016, 73 cm of new snow accumulated in less than 24 hours. This was well captured by the HARMONIE weather prediction model (Driesenaar, 2009), compared to weather radar images.

Severe freezing rain: Climatological information about the occurrence of freezing rain (FZRA) can be produced by applying a precipitation typing algorithm that is based on vertical profiles of relative humidity and temperature (Kämäräinen et al. 2017). Almost regardless of the latitude, the coastal and marine areas do not experience FZRA as often as the other regions (Fig. 2.1.2.1), apparently because warm water bodies effectively prevent the occurrence of near-surface cold layers. An exception is the northern Baltic Sea, having a long ice cover season and relatively frequent FZRA cases. In the future, freezing rain probabilities are projected to increase in northern and decrease in southern Europe.

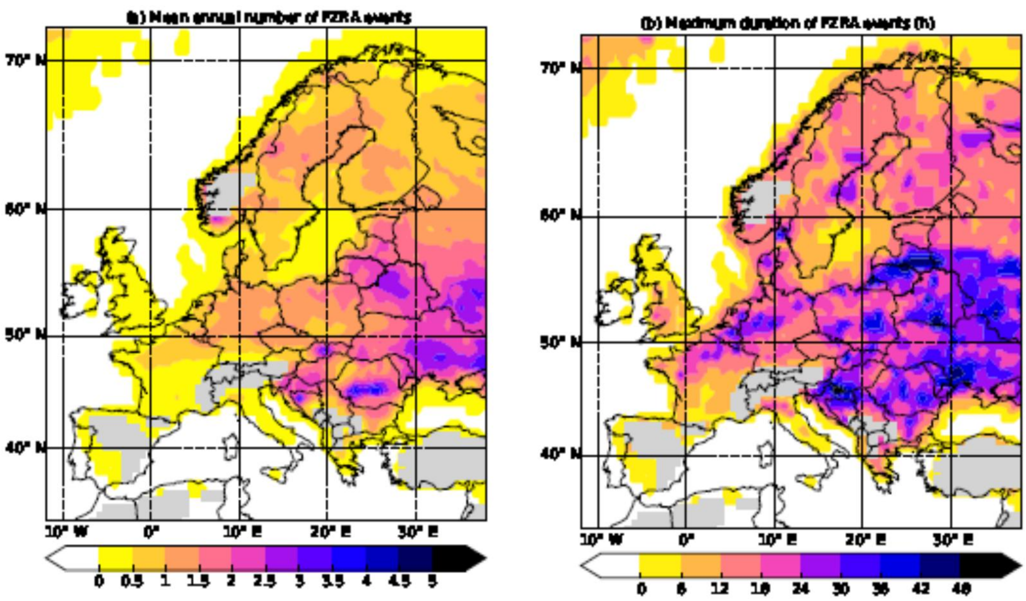


Figure2.1.2.1. (a) Mean annual number of freezing rain events and (b) maximum duration (hours) of the events in the 1979–2014 study period based on vertical profiles of relative humidity and temperature in the ERA-Interim reanalysis data. The highest elevations were excluded (grey) because of large uncertainties there in the method used (Kämäräinen et al. 2017)

Strong winds: Although climate change effects on the windstorms in Finland are rather uncertain, there is new evidence that the average intensity of the most destructive storms has increased by more than a factor of three in Western, Central and Northern Europe after 1990 (Gregow et al. 2017). On the other hand, even if the windstorm climate in Finland remains the same, the impacts of the storms will alter due to the ongoing change in environmental conditions (decrease in soil frost, increase in forest growth), urbanization, and increased dependence on electricity and communication networks.

New insights on the joint effect of sea level and waves: The height to which sea water rises in a flooding situation on a shore is a combination of the effects of both sea level and wind-generated waves. Local wave height conditions vary greatly depending on the shape of the shoreline, topography of the seabed, and archipelago shielding the coastline against largest waves. In 2015, we started developing a method for combining the effects of sea level extremes and wind-generated waves to obtain a probability distribution representing their combined contribution to a flooding event. This work was continued in 2016, resulting in an article draft which will be finalised and submitted for peer-review in 2017. The method was applied to two sites at the archipelago area of Helsinki. In the future, the method will be further developed by studying the dependency between high waves and high sea level. The method can also be applied to NPP sites, provided that, in addition to water level observations and future predictions, sufficient information on local wave conditions is available.

Short-period sea level oscillations: Meteotsunami statistics from 1920s up to 1980s, obtained previously in EXWE, were complemented by analysing the digital sea level recordings at 1-minute time resolution from the Finnish tide gauges from 2004 onwards. The synoptic conditions which created the strongest events were found to differ between the summer and winter seasons. Summer events were generally caused by cold fronts or mesoscale convective systems, accompanied with thunder and rapid air pressure variations. Winter events, on the other hand, were related to cold fronts propagating over the sea, with storm winds frequently present. Sea level oscillations were generally slower than during the summer events, indicating that the events were small storm surges rather than meteotsunamis. All of the winter events occurred in mild ice conditions. Highest recorded oscillations at the tide gauges were about 10–40 cm in both seasons, but locally the oscillations may be considerably stronger due to amplification by coastal topography. The results were presented in an article draft which will be submitted for scientific peer-review in 2017.

Review on sea level rise and ice sheet instability: A literature review on global mean sea level scenarios was performed. The purpose was to ensure whether the previously calculated local mean sea level scenarios for the Finnish coast (e.g. Kahma *et al.* 2014) are still valid or whether an update is needed. The overall uncertainty range of the global mean sea level rise projections for this century has not changed lately: it still extends from about 20 to 200 cm. However, recent research has brought more information on the shape of the probability distribution of sea level rise, as well as on regional factors. These could be used to re-evaluate the method used to calculate the scenarios for Finland, but major changes on the scenarios are not expected. The largest uncertainty in sea level rise predictions is the possible instability of the marine sectors of the Antarctic ice sheet. Recent observations indicate that the instability mechanism is already contributing to ice loss from the West Antarctic Ice Sheet. A worst-case estimate of a sea level rise contribution of more than 1 m from Antarctica was presented by DeConto and Pollard (2016). This could mean a worst-case global mean sea level rise of up to 2.5 m during this century.

Extreme space weather: From the viewpoint of GIC, fast geomagnetic variations have the key role. Large magnetic storms can occur at any time during an approximately 11-year solar cycle, with the highest probability within a few years before and after the sunspot maximum. Spring and autumn are more likely periods for enhanced activity than summer and winter. During a day, the most active times are related to the occurrence of auroras in the evening and around midnight.

Severe GIC events with documented adverse effects on power grids occur somewhere in the world only for a few times in a sunspot cycle. Extrapolations of European geomagnetic data indicate that a 1-in-100 year event could be about twice the value observed within about the previous 20-30 years (Wintoft et al., 2016). A couple of recent studies indicate that events nearly reaching the expected 1-in-100 year extreme have occurred within the latest 40 years. In Finland, no serious effects have occurred, thanks to the robust design of the high-voltage grid. Because large GIC events can still be expected, they must be taken into account when installing new transformers or power lines affecting the GIC distribution across the grid.

Towards an integrated radiation dispersion and dose assessment tool. Meteorological modelling, using e.g. the HARMONIE numerical weather prediction system has progressed to a stage where a proper assessment of coastal meteorological conditions can be fed into the dispersion modelling system. A version of HARMONIE, suitable for providing input for high-resolution dispersion modelling on a kilometre-scale, was set up and executed on a domain of about 300 km x 300 km, with a grid having a spacing of only 500 m in the horizontal. In order to correctly represent the important air-surface interactions, more accurate topographic data needed to be implemented. The system was used to generate hind-casts for April and May 2015. Comparing model data to observations (Jurvanen, 2015) confirmed that HARMONIE responded to the thermal and mechanical contrast between land and sea in a realistic manner.

Deliverables in 2016

- A report and two oral presentations on historical time series of extreme convective weather.
- A scientific journal manuscript about atmospheric conditions and circulation patterns favouring significant hail events in Finland was prepared.
- A manuscript about evaluation of thunderstorm predictors for Finland using reanalyses and neural networks was submitted.
- Two posters were prepared, one about severe sea-effect snowfall on the Finnish coasts and the other about variability and trends of extreme temperatures in Finland.
- A manuscript was prepared and a poster presented about estimates of present-day and future occurrence and amounts of freezing rain in Europe based on CORDEX regional climate models.
- A master thesis and an oral conference presentation on extratropical transition and characteristics of storm Mauri in September 1982.
- A report reviewing strong winds in Northern Europe in the past, current and future climate
- A manuscript and three oral conference presentations on a probabilistic approach for combining sea level variations and wind waves has been prepared.
- A manuscript on short-period sea level oscillations and their meteorological background on the Finnish coast has been prepared.
- A report reviewing recent results on future sea level rise and ice sheet instability
- A report on extreme space weather, and a scientific paper (Wintoft et al., 2016)
- A report was written on high resolution meteorological modelling with HARMONIE, together with a report on preparation of high-resolution physiography data files.

- A technical report on the current user interface for SILAM v5 chemical transport model

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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 the 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.

MAPS project aims at the following specific *objectives*:

(a) Identifying the generic safety principles of managing complex projects in the nuclear industry.

(b) Clarifying 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.

(c) Facilitating 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 2016

First, we focused on validating and refining the developed conceptual governance model to take into account the characteristics of different types of projects and the use of multiple governance approaches in a single project. The research work was conducted through theoretical development, case studies and workshops across the work packages in MAPS. We analysed practical empirical examples in which decision-making and the behaviour of project actors have a significant influence on safety, and how the underlying set of governance approaches influence the project actors' activities. This builds understanding of the practical relevance of project governance approaches with regard to enhancing safety. We also explored the applicability of project alliancing/collaborative contract arrangements to the nuclear industry context and lessons learned from managing complex projects during an international workshop, which we organized. An important specific goal in 2016 was also to proceed further with collecting and analysing empirical material in two projects in the Finnish nuclear industry.

Second, we focused on benchmarking between the nuclear industry and the Norwegian oil & gas industry in terms of governance of safety and management of complex projects. In 2016 the goal was to focus on the oil & gas industry and the nuclear industry experiences and management of complex projects and related innovations. The Finnish nuclear and Norwegian petroleum industries are very different branches, yet they share similar goals, such as continuous improvement of safety. In addition, both industries have faced similar challenges, such as dealing with economically hard times, ageing of personnel and infrastructure, decommissioning, managing complex projects with long supply chains as well as increasing automation and related safety and security concerns. To discuss these issues, a joint cross-industry seminar with representatives of the Norwegian oil & gas industry and the Finnish nuclear industry and regulator was held in 2016.

Third, we continued our research on safety culture in complex network organizations. In 2016 the work focused on modelling the cultural complexity and safety culture challenges in projects, as well as on identifying and specifying methods to improve and facilitate safety culture in complex projects. The goal was to review relevant literature to identify the practical methods to improve and assure safety culture. We also conducted three empirical case studies in Nordic nuclear industry organizations, focusing on a safety culture ambassador group as a method, a safety-oriented project management seminar and information exchange with experts to provide additional insight into the current challenges and opportunities of safety culture work in projects. The aim was to develop a preliminary framework for evaluating the applicability of safety culture assurance and improvement methods was developed based on the theoretical and empirical work.

Forth, we continued our study on the applications of system dynamics modelling in complex projects. In 2016 we focused on discussions with nuclear industry practitioners and representatives of the regulator for providing important practical insights and input for the systems dynamics modelling. We prepared a causal loop diagram to indicate how some of the project governance and safety aspects are related in a nuclear industry project and summarized the findings from 2016 in a report.

The fifth goal was to internally integrate and disseminate the results from the MAPS project. The international visibility of MAPS project results in 2016 was ensured by participation of the researchers in eight different events, such as international scientific conferences, domestic industry seminars, expert group meetings, etc., as follows: IAEA International Conference on Human and Organizational Aspects of Assuring Nuclear Safety – Exploring 30 Years of Safety Culture, European Academy of Management (EURAM), Society for Risk Analysis (SRA), European Safety and Reliability (ESREL), SAFETY2016 World conference, FinNuclear Quality forum (Japan Nuclear Safety Institute, JANSI), International workshop on Relational Contracting for Improved Network Performance in Complex Projects, HUSC Expert Group Meeting on Safety Culture.

Deliverables in 2016

- Master's thesis was finalized focusing on the governance of safety critical projects. The key elements in project governance and their effect on nuclear safety were explored both by a systematic literature review and analysis of empirical data. The objective of the thesis was to evaluate how different project governance mechanisms are related to nuclear safety in a temporary project setting (the context of nuclear industry projects).
- A conference paper was written to increase our understanding of how to build governance systems for nuclear industry projects to enhance safety - we analyzed the influence of governance on achieving safety goals in a complex nuclear industry project.
- An international workshop for the nuclear industry was organized on the topic "Relational contracting for improved network performance in complex projects" to discuss with nuclear industry practitioners, researchers and international academics about management of complex projects and applicability of project alliancing to the nuclear industry context.
- Scientific article manuscript was written to address drivers of complexity in inter-organizational nuclear industry project networks, focusing on the implications for safety. A novel integrated conceptual framework for governance in complex nuclear power industry project networks was proposed, which presents key factors that influence the complexity of such project networks.
- A cross-industry Nordic video seminar was organized with focus on lessons learned on managing safety and complex projects in the petroleum and nuclear industries.
- An article manuscript was written addressing simple rules and lessons learned in the complex project management in the Nordic oil & gas and nuclear industries. The paper examines some critical points in the project management that may cause delays or endarger safety.
- A conference paper was written to provide an overview of the ongoing research activities in MAPS project and highlight the interdisciplinary approach for advancing theoretical and practical understanding on enhancing safety culture and nuclear safety in complex nuclear industry projects.
- A NKS AIM project workshop was held in Stockholm on safety culture improvement methods with MAPS researchers and representatives of the case organizations, focusing on theoretical and empirical discussion.
- The NKS AIM project held a presentation and group work on safety culture improvement methods in the NPP lifecycle at the HUSC Expert Group Meeting on Safety Culture.
- An extended conference abstract was written, presenting a safety leadership approach on building an 'adaptive safety culture' in the nuclear industry.
- A NKS AIM project produced an intermediate report "Safety Culture Assurance and Improvement Methods in Complex Projects"
- A VTT report was written on the topic of delays in creating and handling design documents from a simulation modelling perspective.
- An invited presentation at international nuclear industry forum on the topic of safety culture in complex nuclear industry project networks.

- A conference abstract was written and presented on the topic of inter-organizational complexity and safety culture in nuclear industry projects

2.1.4 PRAMEA - Probabilistic risk assessment method development and applications

The general objective of the PRAMEA project was 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 provided a Nordic review of the use of human reliability analysis outside of PRA, an examination of performance shaping factors when assessing human error probability in advanced control rooms, participated in the preparation of an IAEA safety report for HRA. It has outlined a framework for conducting multi-unit PRA analyses together with a Swedish partner, reviewed the applications of the dynamic flowgraph methodology (DFM), and published a scientific paper on prime implicants in DFM. Further, it has analysed factors that affect release height and temperature in level 2 PRA, implemented a way of tight integration between level 1 and level 2 PRA, wrote a user's guide for new features in the FinPSA code, updated FinPSA design, tested the new features of FinPSA. In level 3 PRA, it has reviewed recently used dose assessment methods in level 3 PRA, participated in the writing of guidelines for level 3 PRA. Finally, it has focused on optimizing risk-informed decisions in safety critical contexts. .

Specific goals in 2016

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.

The objective of HRA outside the PSA was to investigate why HRA is not typically used for evaluating the operators' contribution to safety outside of the PSA, and to identify non-PSA application areas in which HRA could provide valuable insights and inputs.

The objective of the Multi-unit PRA modelling task was to outline a simple method for tentative estimation of multi-unit risks. The work was performed in collaboration with Lloyd's Register Consulting.

The objective of the dynamic flowgraph methodology (DFM) task was to conduct a literature survey focusing on the applications of DFM.

The objective on level 2 integrated deterministic and probabilistic safety analysis (IDPSA) task was to study factors that affect the height and temperature of a radioactive release from a nuclear power plant. A plan was also to continue the development of a simplified level 2 PRA model, and implement uncertainty analysis for probability parameters in the model. In addition, a plan was to prepare a conference paper of previous IDPSA research.

Level 2 method support objectives were knowledge transfer to new developers and renewal of outdated SPSA application. Actions were needed because SPSA does not natively work in computers of today and existing models included important knowledge. In 2016, the objective was to design and implement tight integration between level 1 and level 2. .

VTT has developed in 2014 a pilot to analyse the radiological consequences of the radionuclide release of the Fukushima Daiichi source term of 2011 in the imaginary case that the inhabitants of five major cities close to the NPP site would have been in place (instead of having died in or been evacuated due to the March 2011 earthquake and tsunami) when the release occurred. In 2015, the objective was to improve on this pilot to make it more realistic.

Emergency operations – e.g. recovery from a loss of coolant accident at a nuclear power plant – contain risks (schedule risk, end product quality risk, cost risk, side effect risk, injury risk) that can be analysed by applying methods from project management. In 2015, the objective was to study methods of analysing schedule risks when resources available for the operation are limited. The methods were to be applied to a small case.

The objective of reliability analysis of defense-in-depth in organizations focuses on two topics in relation to defense-in-depth strategies: (i) the assessment of the impact of communication and coordination errors on organizational decision-making processes and thus the implementation of activities in support of safety goals; and (ii) the development of robust strategies for ensuring the safety of systems, especially when there is a need to detect and eliminate errors through cost-effective deployment of inspections.

Deliverables in 2016

- A conference paper Markus Porthin, Marja Liinasuo, Terhi Kling: Recent Development and Future Prospects in Human Reliability Analysis of Advanced Control Rooms in Nuclear Power Plants. Enlarged Halden Programme Group Meeting, Oslo, Norway, 8-13 May, 2016.

- A report examined the use of performance shaping factors (PSFs) in human reliability analysis (HRA) methods when assessing the human error probability (HEP) in post-initiator situations in the main control room, with focus on advanced control rooms.

- Participation in a seminar presenting the results of the NPSAG and SAFIR HRA dependencies project. The seminar and related project reports provide a thorough picture of the state-of-the-practice of the treatment of dependences in today's PSA studies.

- Survey on the Non-PSA applications of HRA was performed. In Finland and Sweden, HRA is used in great majority of cases for PSA purposes only. The lack of broader experience of the area seems to be one reason for that as in the survey, the need for guidance was emphasized. A reasonable way to solve this problem would be to strengthen HRA from the inside of the HRA community. Lots of possibilities for HRA applications were identified.

- Participation in an IAEA expert group to develop a Safety Report on Human Reliability Assessment for Nuclear Installations. The initial meeting of the effort was held in November 2016 in Vienna.

- We have outlined a methodology for preliminary multi-unit PRA. The methodology aims to estimate multi-unit core damage frequencies (or large early release frequencies) related to different multi-unit dependencies. The suggested approach is based on identification of multi-unit dependencies, qualitative and quantitative screening, estimation of frequencies/probabilities for screened in dependencies, and conditional quantification of existing single unit PRA models. The approach is considered conceptual, and there are a number of outstanding issues that could be studied further. Most importantly, the approach needs to be validated using full scope PRA models.

- We performed a literature survey that focused on applications of DFM. The application areas include digital control and safety systems in nuclear power plants, space systems, hydrogen production plants, human performance, networked control systems and field programmable gate arrays. In most of the applications, DFM has been used to analyse how control system failures can cause some physical variable, e.g. water level or pressure, to have too low or high value.

- A journal article on prime implicants in dynamic reliability analysis was published in Reliability Engineering and System Safety. Prime implicants are minimal combinations of events and conditions that cause the system's failure. They can be understood as multi-state and timed minimal cut sets. The paper focused on the mathematical definition of a prime implicant.

- We have studied factors affecting release height and energy in severe accidents, and hydrogen explosions in BWR plant. Release height depends strongly on containment failure mode; only if the failure mode is related to containment wall, there may be some variation in release height. Release temperature is most often 100℃, the temperature of water steam; the temperature may be higher as a result of fires or explosions, and lower if the release route to outside of the reactor building is complex. Inerting of BWR containment has a central role when analysing the risk of a hydrogen explosion inside the containment. Hydrogen explosions can also occur outside the containment and affect releases significantly even though they have not usually been modelled in PRA. Full uncertainty analysis has been developed for a BWR containment event tree model.

- A conference paper on the BWR containment event tree model was presented in 13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13).

- Tight integration of PRA levels 1 and 2 was implemented, and a user guide for it was prepared. There are two main areas in tight integration: how level 1 information is incorporated and utilised in level 2 models; and how level 1 accident sequences and basic events are seen in level 2 results. It can be calculated how level 1 sequences, basic events and initiating events contribute to the values of level 2 variables, namely frequencies of level 2 sequences and release categories, and values of source variables. These contributions represent the importances of level 1 events with regard to level 2 results. This way level 2 results can be traced back to level 1.

- A design document covering tight integration design was prepared and the tight integration feature was implemented into FinPSA. The design document and implementation covered all phases from level 2 model development to computation and result visualisation.

- A test plan and report covering features of tight integration was prepared. Test cases were specified for 10 new properties of FinPSA related to level 1 and 2 tight integration. In addition, usability issues and error situations were taken into account. FinPSA passed all the tests.

- A guidance document on level 3 PSA, suited for Nordic conditions, has been prepared in Nordic cooperation in the L3PSA project. The document addresses the regulatory framework both in the Nordic countries and internationally, international guides and standards, and available software applicable to level 3 analyses. It describes use of level 3 PSA results, risk metrics, resources needed for level 3 PSA, safety criteria, uncertainties, input data, the handling of countermeasures in the analyses, the kinds of consequences that a severe accident may have, and the presentation of results. Conclusions are drawn from the pilot studies and on current level 3 PSA status, and on the need of future work in the field.

- A conference paper on level 3 PRA studies was presented in 13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13). It contained two parts. The first part presented the previous pilot study concerning event tree modelling on level 3 PRA and the Fukushima accident. The second part studied integration of PRA levels 2 and 3.

- Another conference paper in the PSAM13 conference described the guidelines document produced in the L3PSA project.

- A report on population dose assessment in level 3 PRA was written. It focuses on methods used recently either in major studies (the SOARCA study, UNSCEAR study of the Fukushima accident) or as part of modern software (VALMA, SILAM, RODOS). It turned out that relatively little progress has been made in dose assessment methods within the last 20 years. However, dose assessment methods and models could be improved for example by taking the progress in behavioural modelling into account in countermeasures, and the progress in applying Monte Carlo methods to dose assessment in medical physics into account in the analysis of cloudshine and groundshine.

- A scientific paper on risk-based optimization of pipe inspections was published in *Reliability Engineering and System Safety*.

- A paper on an optimization methodology for cost-efficient defense-in-depth strategies has been conditionally accepted by *Reliability Engineering and System Safety*.

- A conference paper on the same topic was presented by Alessandro Mancuso at the ESREL 2016 conference on Risk, Reliability and Safety in September 2016.

A paper on an extension of the optimization methodology accounting for imprecise information has been accepted for ESREL 2017 conference.

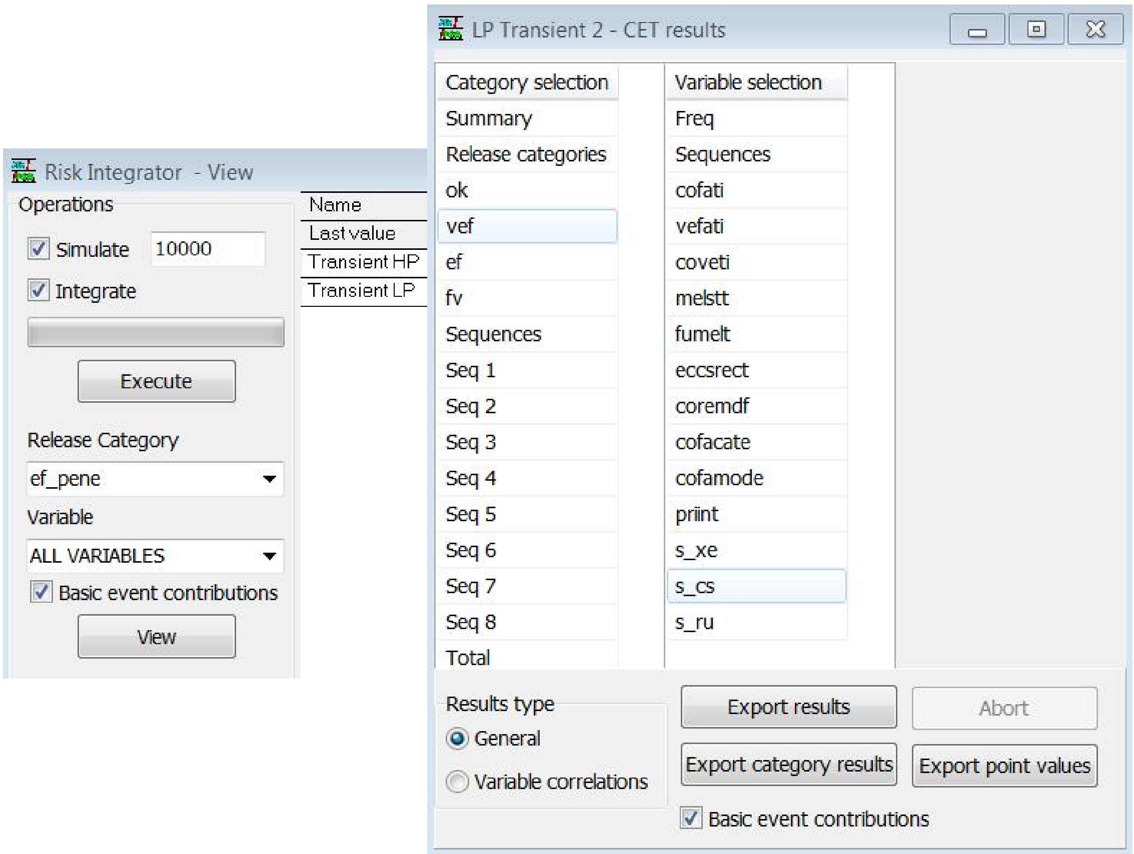


Figure 2.1.4.1. FinPSA Level 2 user interface was changed only slightly due to the tight integration of PSA levels 1 and 2, but the impact of the integration on modeller and analyst productivity is significant.

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

The general objective of SAUNA (2015-2018) is to develop an integrated framework for safety assessment and transparent safety demonstration of nuclear power plant instrumentation and control (I&C) systems. Traditionally, safety assessment tends to focus on technical issues, single faults or limited combinations, deterministic analysis and PRA, and a document-based approach (SARs). In SAUNA, the research theme is overall plant safety. A key challenge is 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 will look 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 will 1) build a shared understanding of the underlying challenges, concepts, and Systems Engineering principles, 2) develop 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) tie the results together into an integrated, structured, model-based approach to safety demonstration and licensing.

Specific goals in 2016

WP1 of SAUNA focuses on clarifying the terminology and Systems Engineering principles in order to provide a common basis for various research activities on NPP safety. In 2016, the focus was on creating a reference model for the qualification process and safety demonstration data, and developing model-based safety assessment methodologies for Defence-in-Depth.

Due to the strict safety requirements, and the long lifetime of NPPs, engineering activities have to be well planned, and their outcomes well managed. Accordingly, SAUNA applies Systems Engineering principles to provide a life cycle management framework. In 2015, an overall model of SE activities was developed in the form of a management plan template (SEMP). In 2016, the goal was to create a reference model for the qualification process and safety demonstration data (Figure 2.1.5.1).

Defence-in-Depth (DiD) is a key issue in NPP safety, but there is a need to reconsider its interpretations and ways of implementing it in practice. In 2015, SAUNA reviewed the concepts and assessment methods related to DiD and I&C architectures. In 2016, the goal was to develop model-based safety assessment methodologies for DiD, focusing on cross-disciplinary models and failure propagation. The aim was to use languages such as UML to describe formal models of power plant and I&C architectures, including, for example, functions, devices, locations and roles of human users.

DiD assessment — based on a probabilistic risk assessment (PRA) model — was also studied in the MODIG (MOdelling of DIGital I&C) subproject. In 2015, MODIG carried out a survey of PRA's role in DiD. In 2016, the goal was to 1) define the scope and needs in the example PRA model for evaluation of DiD capability of I&C systems, based on the requirements of YVL B.1, 2) develop complexity metrics for software reliability assessment, and 3) participate in the OECD/NEA WGRISK meeting as part of a special session on digital I&C.

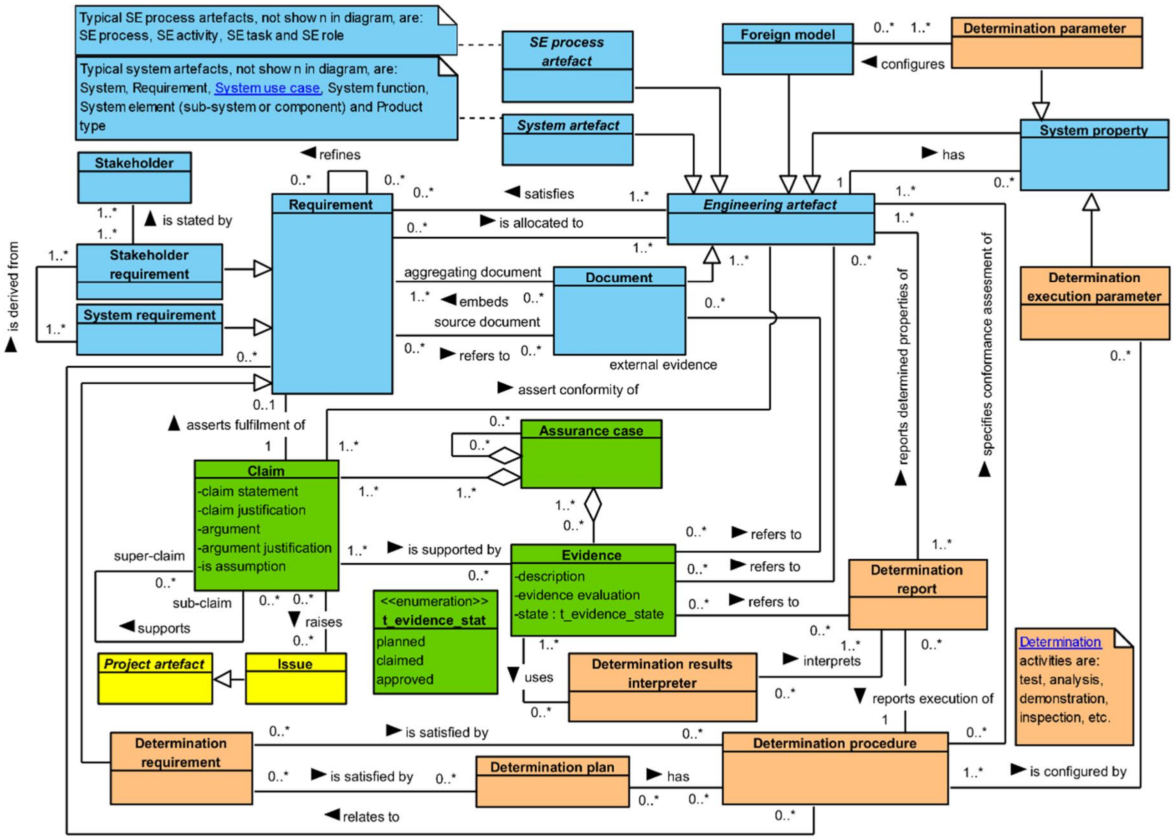


Figure 2.1.5.1. A data model to support model-based safety demonstration with traces to requirements, design artefacts and to the determination results that are used as evidence for the assurance claims.

WP2 of SAUNA looks at the assessment methods and tools for specific technical, human, and systemic safety issues. In 2016, the part of the focus was on integrating formal verification methods (model checking) with simulation models, requirement engineering, and probabilistic reliability assessment. Other points of focus were System-Theoretic Process Analysis, enhancing the Systems Engineering view in the Nuclear SPICE process assessment method, and structured safety demonstration of control room systems.

Model checking is a powerful formal verification method that has been effectively put to practical use in the Finnish nuclear industry. One of the concrete challenges is that the currently used “open-loop” I&C models do not take into account the feedback of the controlled process. A closed-loop approach would filter out unrealistic scenarios, and lead to computationally less complex models. In 2015, an approach for closed-loop analysis was developed and demonstrated with laboratory scale example. In 2016, the objective was to 1) use a generic NPP model provided by Fortum to develop a complete plant-controller model of an NPP I&C system, 2) investigate methods for creating the plant model based on simulation models, and 3) look at the use of over-approximation abstraction in time delay modelling.

Another challenge for the practical application of model checking is that the systems properties to be checked have to be specified in terms of formal languages that are challenging to use, even with some expertise. A domain-specific, user-friendly specification language is therefore needed. In 2016, the objective was to review different visual and textual specification languages and propose a language suited for the nuclear I&C domain. In addition, the aim was to define domain-specific patterns for composing formal specifications.

Model checking and PRA focus of different aspects of plant safety, while still being based on partially similar input information (e.g., the system architecture, hardware configuration, I&C

software design). A unified safety assessment approach calls for better integration of these complementing methods – via a common plant models, or by using the result data from one tool to support the analysis based on the other. In 2015, the feasibility of potential integration approaches was studied. In 2016, the objective was to try some of the integration approaches in practice, based on a simple example system.

System-Theoretic Process Analysis (STPA) is a recently developed method for identifying causes of hazards in safety critical systems, including causes derived from, e.g., societal issues or incorrect control actions. In 2016, the objective was to perform a case study, where the STPA technique is applied in practice (in the context of a Finnish nuclear I&C system), and evaluate the capability of the method in identifying additional safety requirements or constraints, and detecting hazards in system designs.

Process assessment for systems and safety engineering requires development of the current Nuclear SPICE assessment method. In 2015, the focus was in defining requirements for strengthening the model with systems engineering processes according to ISO/IEC 15288. In 2016, the goal was to establish a nuclear domain specific interpretation of process quality that sustains achievement of safety goals in systems engineering. The expected benefit is cost-effectiveness when using process assessments in qualification.

For the multistage validation of control room systems, a Systems Usability Case approach was developed in the previous SAFIR programme. In SAUNA, the focus is on creating a Safety Case based approach for organizing and assessing the fulfilment of Human Factors requirements. An earlier version of such an approach has already been used in the LARA project. In 2016, the goal is to test the approach by applying it to the control room V&V in the ELSA project, compare the results with findings from LARA, and complete the description of the Usability Case process.

WP3 of SAUNA aims 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.

YVL B.1 requires that a qualification process shall demonstrate that systems, structures and components are suitable for their intended use and satisfy relevant safety requirements. Today, the material submitted to the regulator is not always expressed explicitly in terms of arguments and evidence. Accordingly, SAUNA aims at collecting and defining guidelines for structured safety demonstration (Figure 2.1.5.2). In 2016, the objective was to 1) identify the biggest challenges in current practices, based on interviews and analysis of sample material, and 2) define assessment criteria for reviewing whether qualification documents meet the attributes of a good safety demonstration.

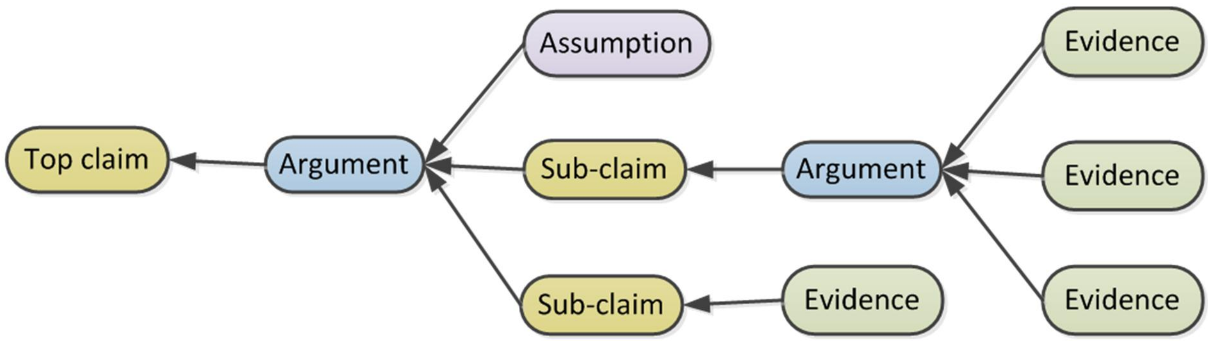


Figure 2.1.5.2. *A structure approach for safety demonstration, where claims are statements about a system's safety properties. Arguments express why the claims are true (or false) based on the sub-claims, evidence items and explicit assumptions.*

Deliverables in 2016

- A research report describing a reference model for the NPP I&C qualification process and safety demonstration data to provide structured Claim-Argument-Evidence cases.
- A conference paper submitted to IDETC/CIE 2017 about a UML based methodology for early multi-disciplinary assessment of DiD capabilities.
- A conference paper in RAMS 2016 on a model-based approach for Human Reliability Analysis during the early phases of Emergency Operating Procedure (EOP) development.
- A conference paper in RAMS 2016 about a cable routing function failure analysis method for early design for protecting the redundancy of critical system functions.
- A research report of the DiD concept with emphasis on level 2 DiD functions (preventive safety functions) and assessment of DiD using PRA.
- A conference paper in PSAM 13 about a complexity measure for analysis of I&C application software.
- A presentation in the annual WGRISK meeting about the DIGREL and MODIG projects.
- A journal paper submitted to IEEE Transactions on Industrial Informatics about a case study of applying closed-loop model checking based on a generic NPP example.
- An accepted journal paper in IEEE Transactions on Industrial Informatics about heuristic construction of a plant model for closed-loop verification.
- A slide set describing an over-approximating approach for modelling of time delays in model checking.
- A presentation in the 9th FPGA Workshop about verifying Field-Programmable Gate Array logics using model checking, based on experience on OL3 systems.
- A doctoral thesis on the use of model checking for verifying NPP safety system designs.
- A research report on user-friendly formalisms for specifying temporal properties.
- A conference paper in ETFA2016 on user-friendly property specification languages, based on property samples collected from practical industry projects.
- Master’s thesis on the application of STPA in the design of safety-critical systems.
- A conference paper in ESREL2016 about the coupling of model checking and PRA.
- A research report about the development of an ISO/IEC 15288 based systems engineering process assessment model and updates of the Nuclear SPICE assessment method.
- A conference paper in QUATIC 2016 about a using the results of a process assessment method as evidence in safety assurance.
- A conference paper in EuroSPI 2016 about situational factors that can be used as a starting point for detailed process and safety assessment.

- A conference paper submitted to ANS NPIC&HMIT 2017 about a Systems Usability Case approach for requirement-based human factors evaluation of complex technical systems.
- A conference paper in AHFE 2016 about an integrated Human Factors Engineering (HFE) process for control room upgrades.
- A conference paper submitted to ANS NPIC&HMIT 2017 about demonstrating and argumenting safety of I&C systems.
- A conference paper in EHPG 2016 about the views on safety demonstration and systems engineering for digital I&C systems.
- Master's thesis on structured safety case tools for NPP I&C systems.

2.1.6 GENXFIN – Safety of new reactor technologies

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

Specific goals in 2016

The Finnish regulatory guides on nuclear safety define the main requirements that a new nuclear power plant must fulfil to be able to be licensed in Finland. In GENXFIN project, the main safety features of selected SMR designs were described and obstacles seen in the licensing process investigated. In addition, special attention was paid to the treatment of passive safety systems. The scope of this investigation was limited to light water reactors because the currently operated reactors in Finland are LWRs in which the regulations are based on.

The Finnish licensing process has been designed with large LWR's in mind. This makes the licencing process quite rigid and does not take into account the different design features of SMRs like modularity and multi reactor installations. This being said, no reason why SMRs could not be licensed to Finland, if the Finnish regulatory demands are met, has been identified. The lack of detailed publicly available information on the SMR designs makes it so, that nothing sure can be said on the licensability of particular SMR concepts.

The DiD principle is the basis of the safety design of the reviewed SMRs and also the foundation of the Finnish regulatory guidelines of nuclear safety. The passive decay heat removal safety systems, featured in many SMRs, are taken into account in Finnish regulations by giving them a reduced failure criterion (N+1) com-pared to (N+2) for active systems.

The main hurdle seen in the licensing of IPWR SMRs is the independency of DiD levels. Also the Finnish regulations demand that there are independent safety systems against severe accidents. The actual licensing process is very time consuming and the public information is in no way detailed enough to truly give a definitive statement on the issue.

The materials used in some light water cooled SMRs, challenges caused to joining techniques and non-destructive inspection by the integral nature of the SMRs, and issues related to the acceptance of new materials to nuclear power plant applications are described here. At present, very little information is publicly available concerning the specific materials used in different LWR SMRs, but in general everything implies that the main materials are austenitic stainless steels 304L, 316L, 347, nickel base or high nickel alloys 690, 800, pressure vessel steels and zirconium based alloys, i.e., the same that are used in present day large LWRs.

Due to the tight spaces resulting from the integral construction on-site welding may be difficult, which emphasizes the need for high reliability of the applied welding process. Along with this, ensuring higher weld quality concomitantly calls for the development and implementation of advanced NDE methods, as well. This can be accomplished, e.g., by robotization of welding fabrication and/or using automated welding processes coupled with in-situ monitoring of welding.

The close-packed structures very likely result in a need to develop new or improved NDE methods. A number of questions concerning the in-service inspections were left open due to the lack of detailed information of the components and the access into them.

In the present review, no implications of the application of new materials were found. However, if new materials are used, they should be included in ASME boiler and pressure vessel or other corresponding code. The acceptance process is not simple and strong co-operation between national/international recognized organizations and ASME or corresponding code administrative committee is needed.

One of the objectives in GENXFIN project 2016 was to participate in the international networks and working groups and increase the interdisciplinary research activities and knowledge transfer within Finland in future reactor technology areas. It is important to participate in development of future reactors in Europe from the start as joining later has smaller prospective. Active following of the European Sustainable Nuclear Industrial Initiative (ESNII) Task Force (TF) enables participation in the coming industrial projects. Participation in the IAEA activities, GIF working groups and European Energy Research Alliance (EERA) Joint Programme Nuclear Materials (JPNM) has enabled VTT to remain in active role in the global nuclear field especially in materials and manufacturing technology of new reactor concepts.

Deliverables in 2016

- Safety Features and Licensing Issues of Integral Pressurized Water Small and Modular Reactors
- Review report on SMR material issues
- EERA JPNM pilot project TASTE: A journal article by R. Pohja et al., “Multiaxial creep testing device for nuclear fuel claddings”
- Travel reports EERA NM & ESNII meetings
- GIF annual report → 1 GIF/IAEA travel reports
- Travel report Energiforsk conference: Nuclear Technology and Policy Developments - a Global Perspective

- Near term strategy plan, The Vision: Nuclear power will be the key to prosperous zero-carbon future

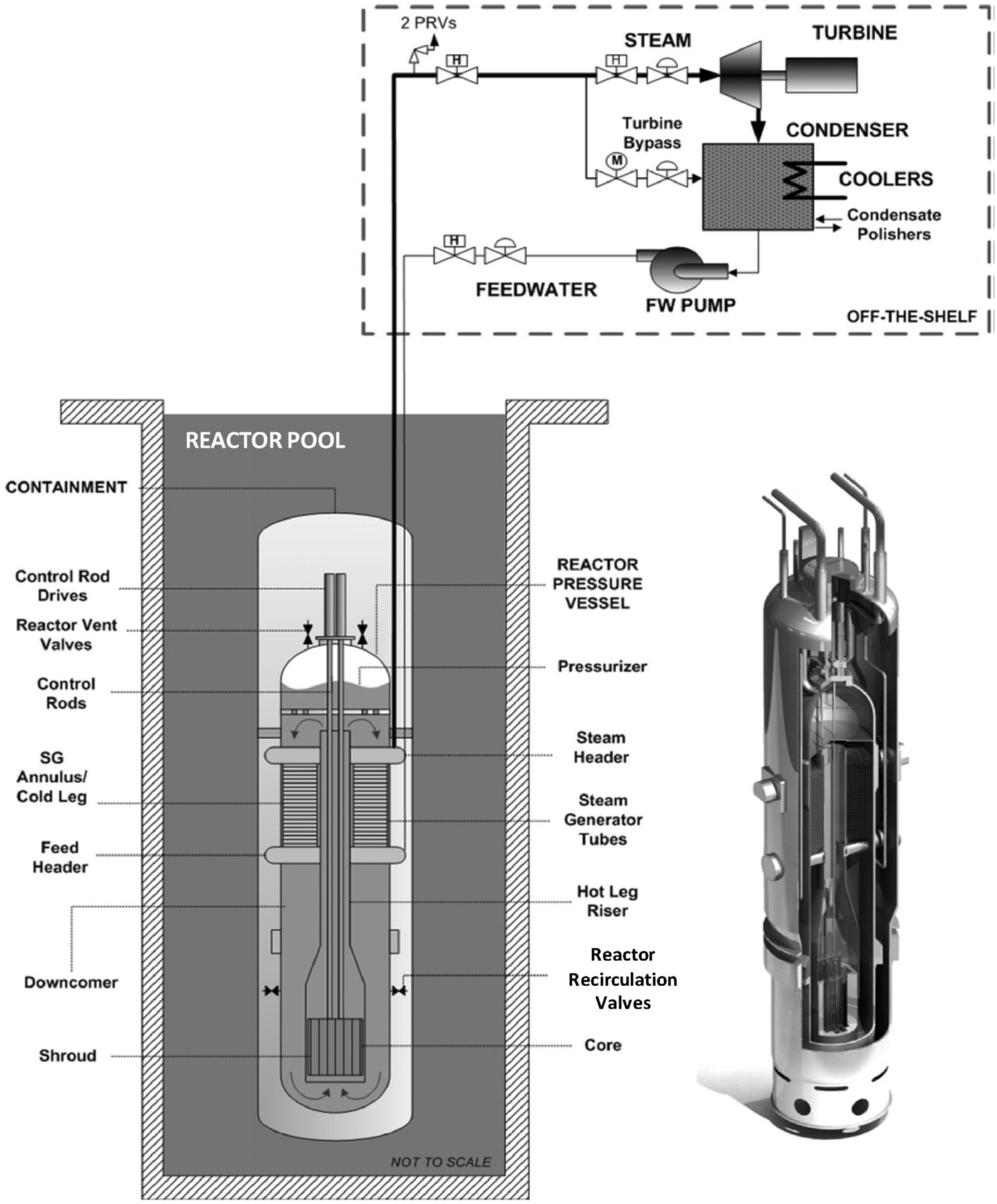


Figure 2.1.6.1. Engineered safety features of NuScale power module. Source: NuScale, 2011. NuScale Plant Safety in Response To Extreme Events. Nucl Tech, 128, 153-163 (May 2012)

2.2 Reactor safety

In 2016 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 2016

Fukushima accident provides a unique opportunity for gaining more information on the progress of severe accidents and their prevention and mitigation. In 2016, the Unit 3 model was updated. The reactor was cooled by the RCIC (Reactor Coolant Isolation Cooling) and HPCI (High-Pressure Coolant Injection) systems for the first 36 hours. The flow rates of these systems were manually adjusted so that the measured water level and pressure in the reactor are reproduced. The rise of the containment pressure accelerated after 6 h. This might be caused by stratification of the wetwell water, but it could not be reproduced by a simple stratification model. A small leak from the recirculation pump seal to the drywell was assumed, starting at 6 h 20 min, in order to reproduce the measured pressure increase.

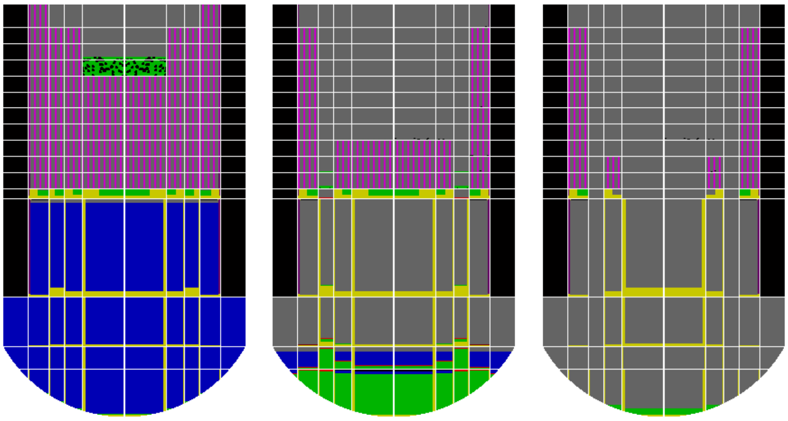


Figure 2.2.1.1. State of the reactor in Unit 3 at three instants of time: 41 h 52 min, 43 h 44 min, and 72 h. The pink colour shows fuel rods, yellow is steel structures, green particulate debris and blue is water.

RPV lower head penetration failure was calculated to occur at 43 h 44 min. It is uncertain whether a failure of an instrument penetration is sufficiently large to cause a discharge of debris out of the reactor, but that is what happened in this calculation. The lower head was cool at this time, and removing the penetrations from the model would delay the RPV failure to a much later time. At the end of the calculation, 31 % of the fuel was still in the reactor, while 69 % of the fuel had been discharged to the containment. The state of the reactor at three instants is illustrated in Figure 2.2.1.1.

Coolability of corium should be ensured in all of its locations and forms. Previously the effect of debris bed geometry and flooding mode on dryout heat flux have been analysed experimentally and analytically to evaluate the coolability of an ex-vessel debris bed. However, the coolability limit based on the minimum dryout heat flux might be overly conservative, since the temperature may remain on an acceptable level even in the dry zone. Instead of the dryout heat flux, it has been proposed that the coolability limit should be based on the increase of the particle temperature. To analyse this, the behaviour of conical debris beds was studied by performing MEWA simulations examining the influence of the bed particle size, heating power and porosity.

The MEWA results were also compared to the KTH's DECOSIM results The simulation results are not in fully agreement. For small particle cases without temperature stabilization, the codes agree satisfactorily. On the other hand, these cases are not interesting because the maximum particle temperature eventually exceeds the temperatures where zirconium oxidation or even corium re-melting begins. In the other conical bed cases, the beds are coolable but MEWA and DECOSIM predict different transient behaviours and final steady-state conditions. Therefore, before trying to quantify the temperature-based dryout criterion, the origin of the significant differences between the MEWA and DECOSIM results needs to be identified.

Corium ex-vessel heat transfer was also evaluated in the form of melt pool by testing the new water ingression and melt eruption models implemented in MELCOR. Seven SSWICS (Small-Scale Water Ingression and Crust Strength) and two CCI (Core-Conrete Interaction) experiments were analysed with three variants of the MELCOR model: (1) old code version 1.8.6 default model that does not take water ingression into account; (2) new code version 2.1 default model that attempts to model water ingression by heat transfer multipliers for the boiling heat transfer coefficient and thermal conductivity and (3) with the new water ingression model.

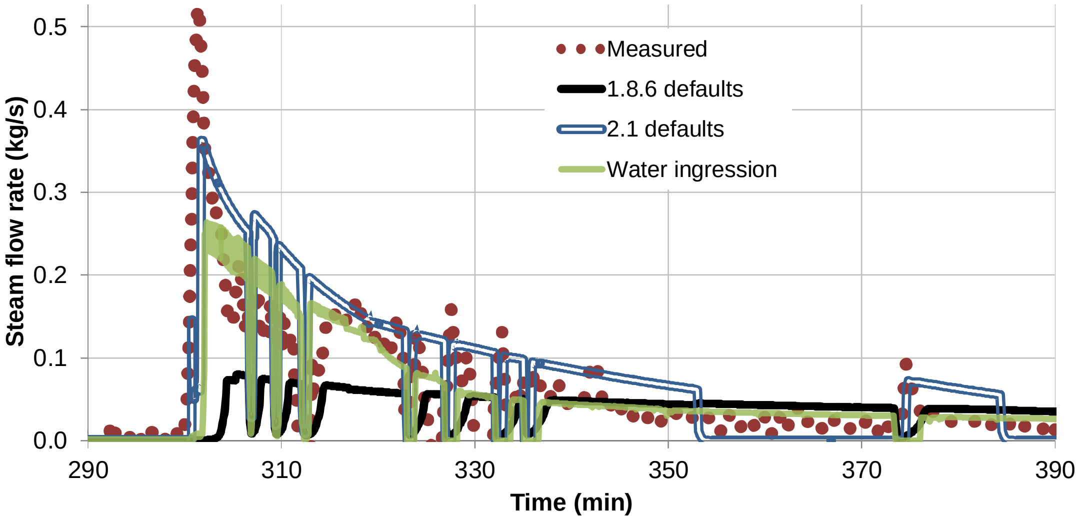


Figure 2.2.1.2. Steam flow rate in the CCI-2 test, measurement compared with three variants of the MELCOR model.

1.8.6 default model underestimated melt pool coolability in seven of the calculated cases as expected. 2.1 default model significantly overestimated the melt pool coolability in all analysed cases. The new water ingression model performed satisfactorily in CCI experiments, in which gas bubbles were released to the melt from decomposing concrete, as illustrated in Figure 2.2.1.2. The new model had little effect in the SSWICS experiments that were done without gas bubbling through the melt.

Accidents that may lead to bypassing the filtered containment venting should be practically eliminated in Nordic Boiling Water Reactors (BWR) that are inerted with nitrogen to avoid hydrogen explosions. However, if the inertion is lost, hydrogen explosions may be possible in the containment. This is most probable during the shutdown or start-up. In the reactor hall hydrogen explosion could occur after the loss of containment integrity. This still has an effect on the timing of the radioactive release and on the quantity also by resuspension of deposited fission products. Hydrogen explosions may also occur in the reactor hall even if the containment is intact if containment pressure evolves to a high level increasing the leak.

The risk of a flammable mixture of hydrogen and air to be formed in the reactor building was studied analysing a Station Blackout (SBO) scenario for the Nordic BWR plant with MELCOR. Without assuming an increase in the containment design leak, the results showed such low concentrations that a hydrogen fire is considered very unlike. The total mass of hydrogen also remained low, so that if the local concentrations could be high enough to be theoretically able to cause a hydrogen fire, the assorted energy release would not be very high and this event could not be considered as an explosion. Also a SBO accident with a non-inerted containment was analysed and this resulted in hydrogen deflagrations in the containment. However, this did not proceed into detonation i.e. into an explosion.

Well-founded dose estimates are needed when assessing the operation of instrumentation and automation systems and containment penetration seal materials under severe accident conditions. Dose rate affects also the formation of nitric acid (HNO_3) in the containment that reduces the pool pH decreasing iodine retention in pools. In-containment dose rates produced with integral code ASTEC and by NRC method were compared. The basic assumption was NRC method would produce higher values than ASTEC because the deposited fission products were included in the gas phase inventory and because ASTEC assumes that the wall absorbs 50 % of the radiation from the deposited fission products. The difference was higher than expected for all but drywell beta dose rate.

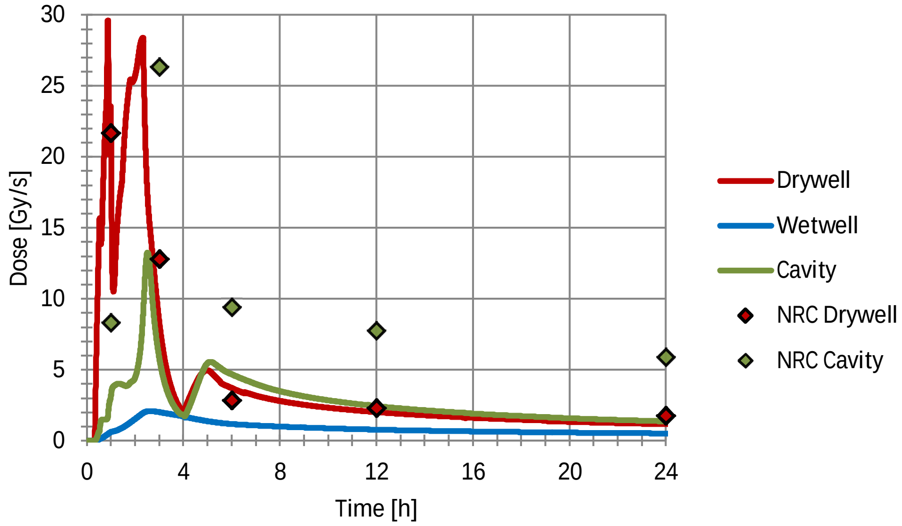


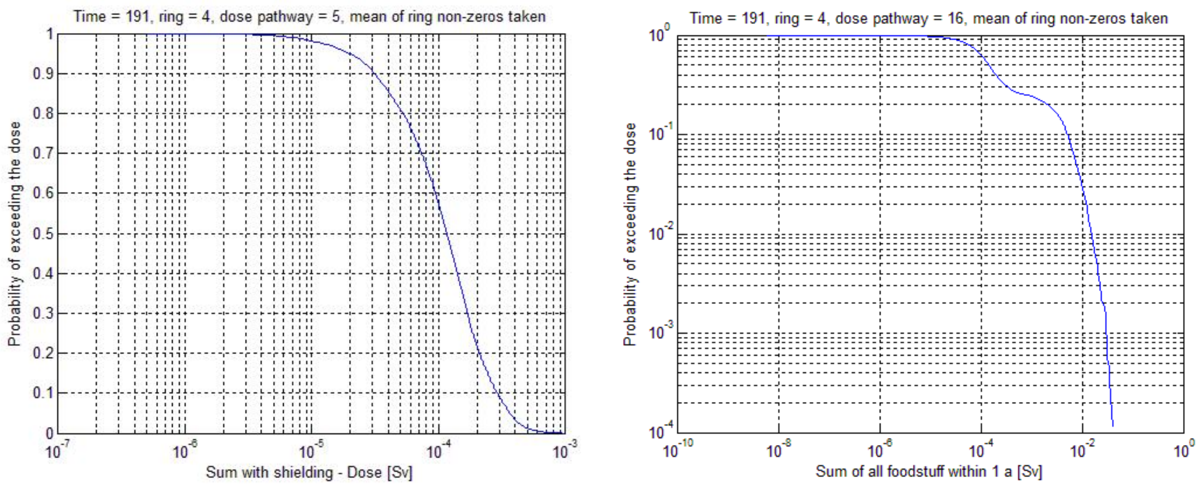
Figure 2.2.1.3. Comparison of total dose rates in the gas phase calculated with ASTEC and NRC method.

As seen from Figure 2.2.1.3, NRC beta dose rate in the cavity is approximately twice the ASTEC dose rate in cavity atmosphere but in drywell they are more or less equal. In ASTEC, the beta dose rate is inversely proportional to the volume of the zone. It is assumed that ASTEC does not take into account the decreasing gas phase volume due to cavity flooding when defining the dose rate. With NRC method, the dose peaks are difficult to observe, due to few data points.

ASTEC input was also changed by increasing the wet painted wall area in the containment. It was expected that this would increase the mass of organic iodide, but it increased more notably the mass of gaseous iodine and iodine oxides. This is assumed to result from organic iodides radiolytically destructing into gaseous iodine that then reacts with air radiolysis products to form iodine oxides. The change in iodine behaviour resulted slightly higher dose peaks in the containment gas phase but notably smaller dose rates on walls.

As a consequence of Fukushima accident IAEA started to develop recommendations which consider emergency planning outside the protection and emergency planning zones. Therefore studies of the expected doses beyond 20 km are needed. Probability distributions of radiation doses from different exposure pathways at distances up to 300 km were determined using three different release magnitudes.

Dispersion and dose assessment code VALMA purposed to serve as an emergency preparedness tool was further developed by implementing there the ingestion dose pathways based on coefficients acquired from the AGRID nutrition dose model. Ingestion dose is important when the release includes notably iodine and caesium. If deposition occurs during a growing season, potential doses from contaminated foodstuffs may be significant. The difference between summertime and wintertime doses can be orders of magnitude. In Figure 2.2.1.4 is compared the mean non-zero values of the total non-ingestion dose and the ingestion dose at the distance of 100 km.



(a) (b)

Figure 2.2.1.4. Mean non-zero value of the total dose at distance of 100 km from the severe accident release limit (a) non-ingestion and (b) ingestion. Exposure / ingestion of one year.

The total ingestion dose was approximately 20 times higher than the dose from non-ingestion pathways, when the integration / consumption time period is one year for both. Ingestion dose remains under 10 mSv with the probability level of 95 %, which indicates that countermeasures on food consumption would not be necessary at the distance of 100 km in the case of the release magnitude corresponding to the severe accident limit value 100 TBq of Cs^{137}. The total dose remained little below 100 mSv at the distance of 20 km which results in the similar conclusion that the prevailing release limit is reasonable.

Deliverables in 2016

- Updates to the Fukushima Unit 3 model were described in a research report. Adding the lower head penetrations caused the loss of reactor pressure vessel integrity notably earlier.

- A travel report from the CSARP/MCAP meeting did summarize the most interesting presentations.

- The coolability of a conical bed could be estimated less conservatively by establishing a temperature-based coolability criterion. VTT's MEWA results on debris bed post-dryout temperature behaviour were compared to KTH's DECOSIM results in a research report.

- The new water ingression model in MELCOR produced good results when gas is bubbling through the melt pool. The validation results are presented in a scientific journal article.

- Hydrogen explosions in the Nordic BWR containments were proven to be very unlike according to the written research report.

- A travel report from the THAI-3 meeting summarized the status of the project including performed and planned experiments as well as analytical activities.

- In-containment dose rates calculated with the ASTEC code and NRC methods were compared in a research report. In all cases, the total dose rate estimates were within a factor of two that can be considered rather acceptable.

- It is important to include ingestion dose to offsite dose assessment because its contribution dominates the total dose. Probability distributions of radiation doses from different exposure pathways at distances up to 300 km using three different release magnitudes are presented in a research report.

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 EXSI-PC 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. 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.

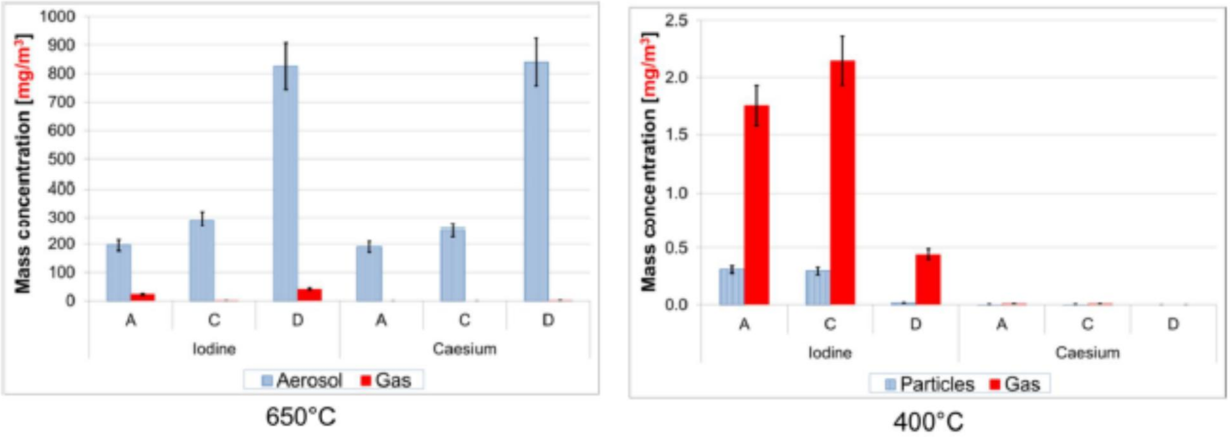
International collaboration is also conducted by participation in the work of OECD/NEA STEM-2, OECD/NEA BIP-3 (both started in 2016) and NUGENIA programmes. The data of experiments performed as part of SAFIR2018 will also be shared within these forums, as well as information related to the progress of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

Specific goals in 2016

The main goal in 2016 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 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.

Another goal was to start 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.



(a) (b)

Figure 2.2.2.1. The transported mass concentrations [mg/m^3] of iodine and cesium in gaseous and aerosol forms under A: Ar/H_2O, C: Ar/H_2O/H_2 and D: Ar/Air atmospheres. When the CsI precursor was heated to 650 °C (a), iodine was mainly transported as aerosol. The decrease of reaction temperature to 400 °C (b) decreased the release of precursor, but the main element released was iodine and it was mainly in the gaseous form.

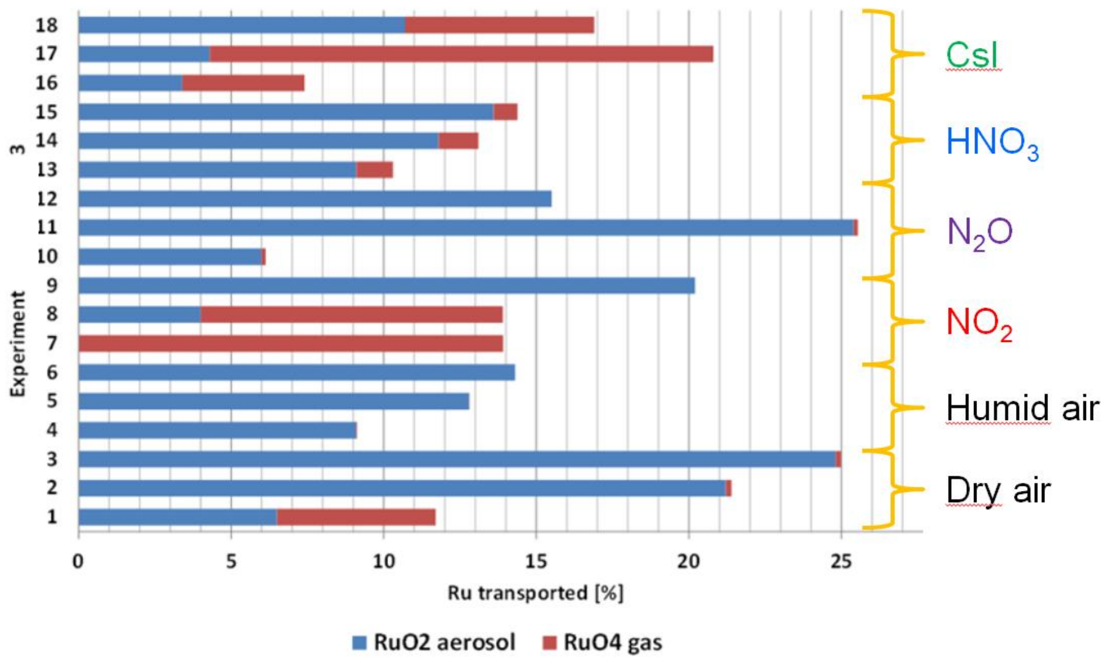


Figure 2.2.2.2. The transport of ruthenium through a model primary circuit was influenced by the composition of atmosphere at 1300 K, 1500 K and 1700 K. The oxidizing air radiolysis product NO_2 increased the fraction of gaseous ruthenium compound, such as RuO_4. The feed of CsI compound seemed to increase the transport of gaseous Ru compound even more than the feed of NO_2. In these experiments, the observed transport of gaseous Ru compound seemed to be even several orders of magnitude higher than in the previous experiments with only pure Ru oxides in the dry/humid air atmospheres. This is not considered in the current severe accident analysis codes due to the lack of experimental data. Other tested additives in the experiments were e.g. N_2O and HNO_3. Nitric acid seemed to increase the transport of gaseous ruthenium compound at all studied temperatures, however the concentration of HNO_3 fed into the facility was lower than for NO_2 and N_2O.

Deliverables in 2016

- In the primary circuit studies the source of iodine was CsI powder which was evaporated at 400 °C on ceramic surface under Ar/H_2O, Ar/H_2O/H_2 and Ar/Air atmospheres. The surface of the reaction furnace tube, made of stainless steel, was pre-oxidized before the experiments. Several mixtures of CsI with B_2O_3 or B_2O_3 + CsOH additives, simulating boric acid dissolved in the primary coolant in Pressurized Water Reactor and to increase the ratio of Cs/I, have been tested. To summarize the main outcome of experiments, a notable release of gaseous iodine from CsI powder was observed at 400 °C, whereas in the previous experiments at 650 °C the aerosol fraction was dominating the release, see Figure 2.2.2.1. The addition of boron released higher gaseous iodine fraction compared to the vaporisation of only caesium iodide under the same conditions. A scientific publication was written.
- The continuation of previous studies in 2014-2015 verified, that the transport of ruthenium through a model primary circuit was influenced by the composition of atmosphere, see Figure 2.2.2.2. The oxidizing air radiolysis products, especially NO_2, seemed to increase the fraction of gaseous ruthenium compound significantly. The feed of CsI compound increased the transport of gaseous Ru compound even more than the feed of NO_2. In these experiments, the observed transport of gaseous Ru compound seemed to be even several orders of magnitude higher than in the previous experiments with only pure Ru oxides in the dry/humid air atmospheres. This is not considered in the current severe accident analysis codes due to the lack of experimental data. Other studied additives were N_2O, HNO_3, Ag and $AgNO_3$. All these findings were summarized in three scientific publications (two of them are listed in this report). A PhD thesis including the publications was finalized in 2016.
- The formation of nitric acid by beta irradiation in humid air simulating containment conditions in a severe accident was verified. The G-value for nitric acid formation was derived. The gathered data was utilized in the Chempool analysis of containment pool pH. As a result, the nitric acid formed in humid air by beta radiation decreased the pool pH. It was concluded, that the effect of beta radiation on the containment chemistry should be studied in detail and also that the amount of NaOH needed for pH control of water pools can be higher than previously expected.
- The experiments on primary circuit chemistry of iodine, performed as part of the previous and current SAFIR programmes, have been noticed internationally and thus a joint project with a Japanese JAEA (Japan Atomic Energy Agency) organisation was started in 2016. The aim is to focus on the effect of boron on iodine chemistry in a RCS. The collaboration includes researcher mobility between the organisations.
- The experiments on primary circuit chemistry of ruthenium, performed as part of the previous and current SAFIR programmes, were noticed in the OECD/NEA STEM-2 project. The new findings on ruthenium chemistry obtained in the SAFIR programmes will be verified in the STEM-2 project and also complementary experiments will be performed.
- The results of iodine and ruthenium studies obtained in the SAFIR programmes (previous CHEMPC and TRAFI projects and current CATFIS project) have also been presented in the ICAPP2016 and NENE2016 conferences.
- Networking with other organisations through NUGENIA has been performed and further joint studies are being expected.
- A scientific publication on the “The Latest Results from Source Term Research: Overview and Outlook”, covering the recent understanding on source term phenomena, was prepared together with the NUGENIA TA2.4 area coordination team.

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

The COVA project aims at developing and promoting a rigorous and systematic approach to the procedures utilized in validation of independent nuclear safety analysis tools. The process enhances the expertise in thermal hydraulic area of Generation II and III LWR reactors and includes as an essential part training of new experts to this relevant area of reactor safety. Main part of the work is 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 forms an essential part of the project: experimental data produced in these activities is directly utilized in the validation work carried out within COVA, and on the other hand, these validation activities support conduction of the experiments, in addition to promoting international cooperation and networking in the field of nuclear safety research.

COVA is 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 is 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 with proper quantification of output uncertainties. Third work package contains 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 includes the participation fees of international research projects and nothing else.

Specific goals in 2016

Assessment of Apros' validation base was performed in the first year of COVA and in 2016 analyses were performed in order to fill the validation gaps identified in the TH and containment model's assessment reports.

A re-evaluation of the FLECHT SEASET reflooding test 32013 has been performed. The calculation results suggests that a new counter-current flow limitation correlation, added in Apros, in addition to an optimized uncertainty coefficient that governs additional heat flux in the vicinity of the quench front can improve the overall prediction in quench front propagation. Some other parameters, such as carryout liquid mass and gas temperatures still exhibit large discrepancies against experimental results, and require further investigation.

Apros reflood models have been tested against a ACHILLES natural reflood experiment (ISP-25). In general the agreement between the measured data and Apros calculations were reasonable. The quench front progression agreed very well with the measurements but the predicted maximum temperature of the rods, and the amount of entrained water were underestimated by Apros.

ISP-7 case of the tubular ERSEC reflooding experiments has been recalculated with the current development version of Apros in order to verify some recent improvements made into the code, and to help identifying aspects in the code that still need to be worked on.

Test T201 performed at the TOSQAN facility has been calculated with Apros containment. The results indicate that the Apros wall condensation and sump evaporation models work well and are able to reproduce the most important phenomena of the test.

In work package 2 new experiment were calculated. These were the following:

OECD/ATLAS A5.1 SBLOCA benchmark was calculated in post-test phase. Steam generator inventory, break flow modelling and steam generator depresussurizaton valve timings were slightly adjusted. These changes lead to a corrected timespan of the benchmark exercise.

ATLAS A5.2 13 % IBLOCA experiment has been simulated with Apros in pre-test phase. For the simulation primary circuit volume was slightly adjusted and core nodalization is enhanced by dividing the axial heat structures into 220 levels. Slight core heat-up occurred until SIT injections quenched the core. Maximum cladding temperature in the simulation was 387 ºC.

Three scoping calculation cases were simulated to give an estimate whether a circulation flow is established between the PANDA facility's vessels. The main phenomenon studied by these scoping calculations was the circulation flow between the PANDA vessels. The LP approach of Apros is able to reproduce the same kind of circulation flow as the more novel CFD like approach of GOTHIC. The distribution of the helium is also quite similarly predicted by both programs.

FONESYS FO-02 extended critical flow benchmark was calculated and the results were submitted. UCL steady-state cases presented a type of problem which is not covered in Apros validation.

Five international cooperation programmes were followed in COVA project. These were OECD/NEA ATLAS, HYMERES and WGAMA, USNRC CAMP and FONESYS network. A scientist participated in the THICKET-4 knowledge transfer seminar.

Participation fees were paid for OECD HYMERES & ATLAS and USNRC CAMP.

Deliverables in 2016

- FLECHT SEASET reflooding test 32013 was recalculated with a current version of Apros
- ACHILLES natural reflood experiment was calculated with Apros.
- ERSEC ISP-7 reflooding experiment was calculated with Apros.
- Wall condensation and sump evaporation TOSQAN test T201 was calculated using the Apros containment code.
- ATLAS A5.1 SBLOCA benchmark was calculated in post-test phase.
- ATLAS A5.2 IBLOCA was calculated in pre-test phase.
- HYMERES HP6 scoping calculations were made with Apros containment.
- Extended FONESYS FO-02 critical flow benchmark was calculated with Apros and the results were submitted to organizers.

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 test facility 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 2016

Specific goals in 2016 included a test series with a SRV sparger in the PPOOLEX facility on the behaviour of a thermocline. Mixing efficiency due to water injection through an RHR nozzle was also studied in a three test series in PPOOLEX. Furthermore, preliminary spray injection tests, where mixing of a stratified wetwell pool by spray injection from above was studied, were carried out. 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. In the CFD calculations task the extensive database gathered in the previous PPOOLEX studies was utilized at LUT by performing CFD simulations of direct contact condensation (DCC) tests with the NEPTUNE_CFD code.

The pressure suppression pool in a BWR serves as a primary heat sink during a loss of coolant accident (LOCA) or when 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 EHS and EMS models and implementing them in GOTHIC code. 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 the 2016 sparger experiment series, the first test was done with all the 32 injection holes at the sparger head open whereas the second one was done with only the eight holes in the bottom row open. Particularly the behaviour of the thermocline between the cold and warm water volumes and the progression of the erosion process was of interest. For this purpose also PIV measurements were tried during the tests.

In both tests the initial uniform temperature profile first changed to a stratified situation and eventually back to a uniform and mixed situation. Along the tests the thermocline moved downwards. Elevation of the thermocline at the end of the stratification phase was predicted well in the pre-test simulations with GOTHIC by KTH when all the injection holes of the sparger head were open. However, the thickness of the thermocline was larger than expected. Complete mixing of the water pool through an erosion process was achieved with quite a small steam mass flow rate in the test with most of the injection holes blocked (Figure 2.2.4.1).

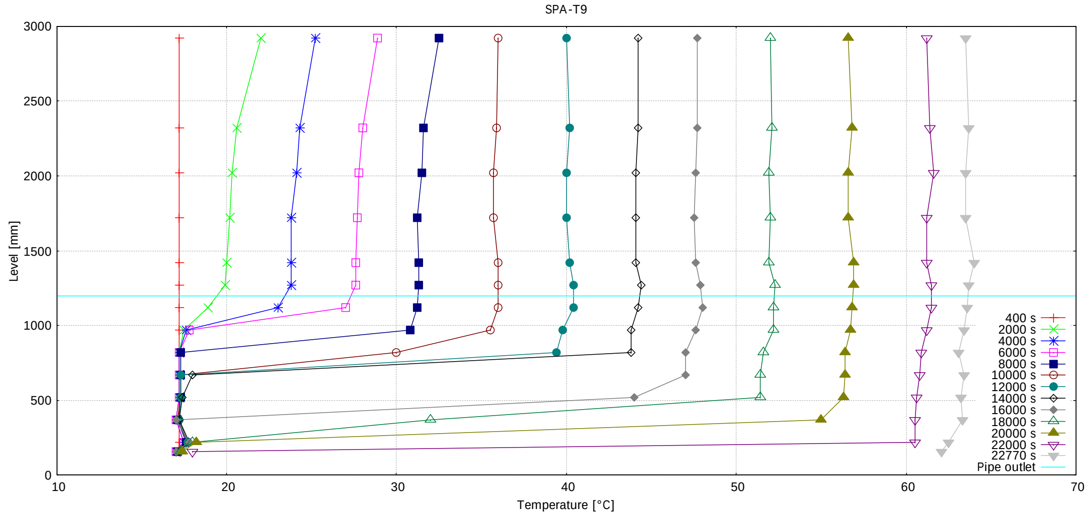


Figure 2.2.4.1. Development of vertical temperature profile of pool water in a sparger test with part of the injection holes blocked.

Recognized flow patterns from the PIV measurements indicate that some kind of swirls could exist around the elevation of the thermocline. Figure 2.2.4.2 shows a velocity vector field averaged over such a 5.7 second time period where the flow patterns were constant enough for the PIV measurement to succeed. Generally, the somewhat chaotic nature of the investigated phenomenon created problems when measuring with a slow-speed PIV system and therefore definitive conclusions on the detailed behaviour of the flow fields in the vicinity of the thermocline can't be made.

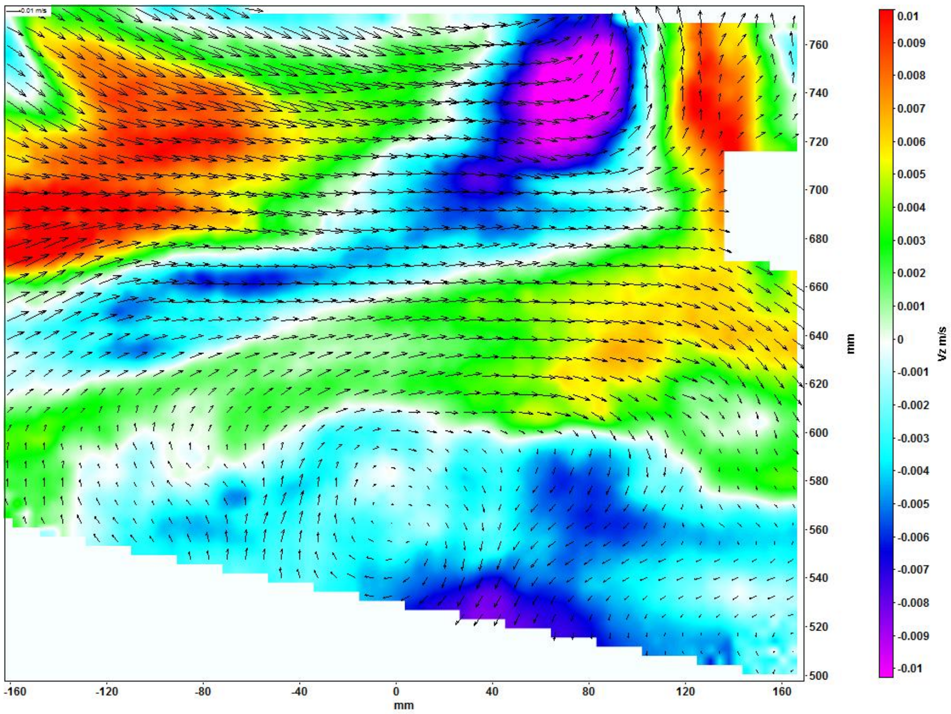


Figure 2.2.4.2. Averaged velocity vector field over a 5.7 second time period from the vicinity of the thermocline.

The mixing mechanism in the SRV sparger tests in 2016 was somewhat different than in many previous tests done in the PPOOLEX facility either with a straight blowdown pipe or with a sparger pipe. Now, the layers of cold water slowly eroded rather than mixed through internal circulation as has been the case in most of the tests carried out before. As a result, the thermocline region shifted slowly downwards as the mixing process proceeded. These tests in PPOOLEX verified that mixing of a thermally stratified water pool can happen through an erosion process instead of internal circulation if suitable flow conditions in the pool created by steam jets at the injection holes of the sparger prevail.

- In 2016, the PPOOLEX facility was equipped with a model of an RHR nozzle. Mixing of a thermally stratified pool with the help of water injection through an RHR nozzle was studied in the tests. Particularly the effects of nozzle orientation, ΔT in the pool, injection water temperature and injection water mass flow rate were of interest. The detailed test specifications were put together and the test parameters were selected on the basis of pre-test simulations with GOTHIC code by KTH. Two stratification and two mixing phases were included. Thermally stratified condition was created by injecting steam into the pool water via the sparger pipe.

- During the stratification phases two regions with clearly different water temperatures and a narrow thermocline region between them developed in the pool. The mixing process was initiated when the target temperature difference between the bottom and the top layer of the pool had been reached. With the vertical orientation of the RHR nozzle mixing was otherwise successful but incomplete above the nozzle elevation (Figure 2.2.4.3). Compete mixing was achieved with the horizontal orientation of the RHR nozzle.

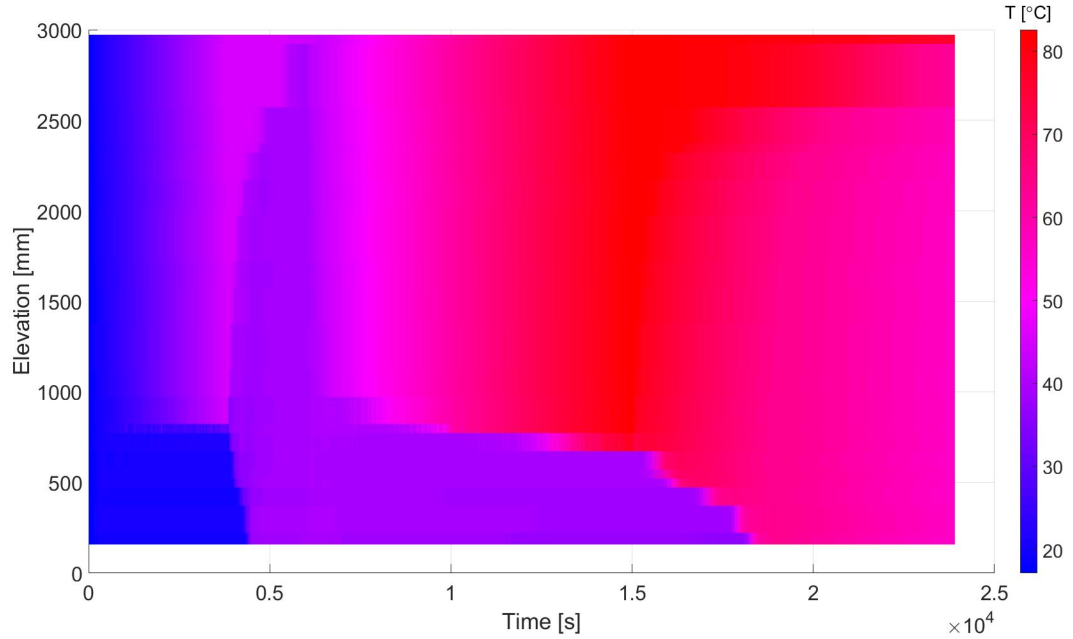


Figure 2.2.4.3. Incomplete mixing in wetwell pool above RHR nozzle elevation (2.5 m) with nozzle in vertical position.

- A four nozzle spray system was installed to the PPOOLEX facility in 2016 and three preliminary spray injection tests, where mixing of a stratified pool by spray injection from above was studied, were carried out. It has been suggested that mixing induced by spray had a role in the pressure drop in Fukushima Unit 3 where pressure build-up in the containment during the first 20 hours after station blackout was attributed to stratification in the pool. Furthermore, the results of these preliminary and forthcoming spray tests in PPOOLEX are aimed to be used for improving simulation models in CFD and system codes.

- The same kind of full cone spray nozzles, which were tested in a separate testing station with the shadowgraphy application in 2015, were used in the PPOOLEX spray tests. These preliminary spray tests indicate that it might be possible to mix a stratified pool with the help of spray injection from above. If spray injection was continued long enough internal circulation developed and finally mixed the pool. Figure 2.2.4.4 shows the development of the stratified situation until 13400 s into the experiment and then the mixing period until complete mixing. The used spray flow rate was about 32 l/min for each spray nozzle and the temperature of spray water about 10 °C.

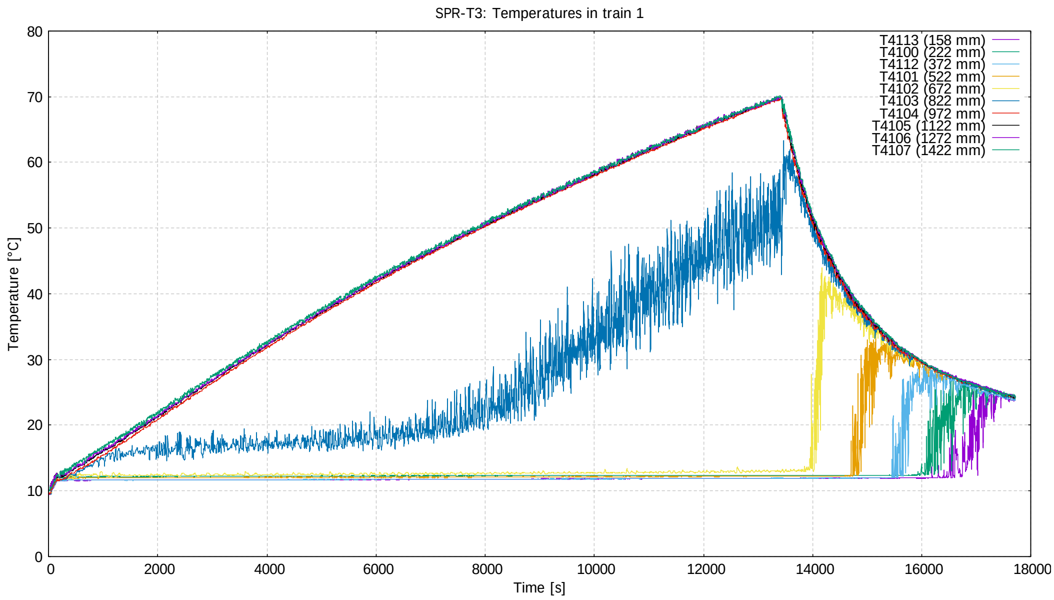


Figure 2.2.4.4. Mixing of a thermally stratified pool with the help of spray injection from above.

- CFD modelling of pressurizing two-compartment suppression pool requires that the interfacial area density between the liquid and vapour phases is resolved either by using a very dense computational grid or by applying a special interfacial instability model. To solve the problem interfacial area density modelling has been addressed by including the effect of interfacial instabilities e.g. Rayleigh-Taylor instability to the NEPTUNE_CFD code. A plausible and simple solution for addressing interfacial area density modelling was introduced in the NURETH-16 conference by Pellegrini et al. Implementation of the model to the NEPTUNE_CFD code has been done and it seems to perform qualitatively well enough.

- Simulations of a plexiglass blowdown pipe case in PPOOLEX have been done in order to investigate the effect of the RTI model on calculation results of direct contact condensation. The transparent plexiglass blowdown pipes allowed the visual observation of chugging inside the blowdown pipes. Low thermal conductivity of plexiglass made it possible to exclude the effect of wall condensation as well. The condensation rate was lower and qualitative characteristics of condensing bubbles were different in the case without the Rayleigh-Taylor instability model compared to the case with it (Figure 2.2.4.5). In general the RTI model seems to give results closer to reality with a low resolution mesh.

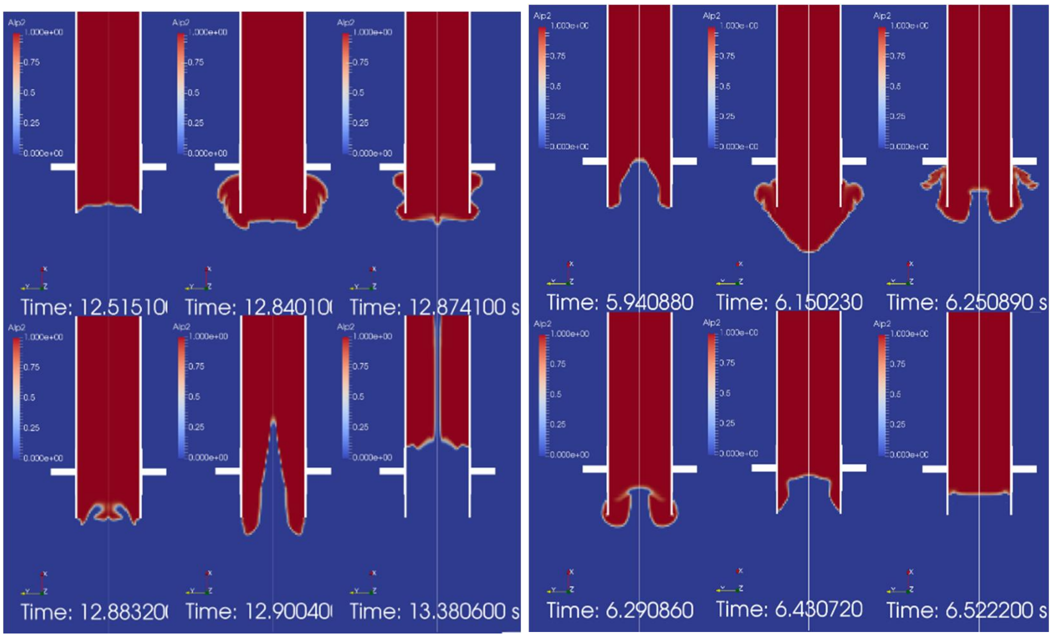


Figure 2.2.4.5. Volume fraction fields from NEPTUNE_CFD simulation of the transparent pipe test with the Rayleigh-Taylor instability model on (left) and off (right).

Deliverables in 2016

- A series of sparger tests on the behaviour of a thermocline was carried out in the PPOOLEX test facility. The elevation of the thermocline moved downwards as the test proceeded. On the basis of PIV measurements it can be concluded that swirls existed at the vicinity of the thermocline. Mixing of the pool was through an erosion process.

- Mixing of a thermally stratified pool with the help of water injection through an RHR nozzle was studied in PPOOLEX. With the vertical orientation of the RHR nozzle mixing was otherwise successful but incomplete above the nozzle elevation. Compete mixing was achieved with the horizontal orientation of the RHR nozzle.

- Mixing of a stratified pool by spray injection from above was studied in test series in PPOOLEX. These preliminary spray tests indicate that it might be possible to mix a stratified pool with the help of spray injection.

- Simulations of a plexiglass blowdown pipe case in PPOOLEX have been done in order to investigate the effect of the Rayleigh-Taylor instability model on calculation results of direct contact condensation. With a low resolution mesh the results seem to be closer to reality when the RTI model is on than when the model is off.

- The development and validation work of the Effective Heat Source and Effective Momentum Source models at KTH has been supported. The validity of the models is being extended to spargers, RHR nozzles and sprays on the basis of the results of the PPOLEX tests.

- Master's thesis on spray droplet size distribution measurement has been published at Lappeenranta University of Technology.

- Journal article on direct contact condensation modeling in pressure suppression pool system has been 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 is to improve the understanding of thermal-hydraulic system behaviour by performing integral and separate effects tests, in particular regarding the impact of non-condensable gases on core cooling and reliability of natural circulation loop decay heat removal. Carefully designed experiments are the most reliable way to obtain fundamental understanding and reliable data of the phenomena. This data will be used in the development and validation of computer codes for the safety analyses of nuclear power plants. 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.

LUT participated in the OECD/NEA PKL Phase 3 project with PWR PACTEL experiments. The project ended in April 2016 and the new OECD/NEA PKL Phase 4 project began in 2016, what Finland is participating in with two PWR PACTEL experiments. The OECD/NEA PKL Phase 3 project was performed with the financial support of the Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2014 and SAFIR2018), the Finnish power company Teollisuuden Voima Oy (TVO), and the partners participating in the OECD/NEA PKL Phase 3 project. The authors are grateful for their support to OECD Nuclear Energy Agency (NEA), the members of the SAFIR2014 and SAFIR2018 Reference Group 4 and the members of the Program Review Group and the Management Board of the OECD/NEA PKL Phase 3 project. The data from the experiments in the OECD/NEA PKL Phase 3 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 2016

Specific goals in 2016 were to make the needed modifications to PWR PACTEL to study nitrogen effects, to perform a test series, and to design a test system to investigate the fundamentals of the selected open natural circulation type heat removal system.

Non-condensable gases, if present in the reactor cooling system, affect the functioning of many safety systems in a nuclear power plant. Nitrogen can temporarily increase the water level in the core by a piston effect. In the INTEGRA project the effect of nitrogen in LOCA situations was studied experimentally with the PWR PACTEL facility. In these tests with PWR PACTEL the safety functions were the accumulator injection, secondary side depressurization, and core power reduction. Four of these tests included the accumulator injection to a cold leg like in the EPR and two to the upper plenum as in the VVER and AES type power plants. The experiments complement needs to have experiment data for code validation.

Many currently marketed LWR designs feature varying numbers of naturally circulating decay heat removal loops. Large diversity of design configurations is available. At Lappeenranta University of Technology (LUT), tentative studies have been on-going on the fundamentals and operation modes of passive heat removal systems. The selected passive system to be studied in more detail was chosen to be an open passive heat removal system. The main emphasis is to study the mechanisms that can endanger the functioning of the passive system.

At LUT the plan is to build a model of an open type passive heat removal loop, with open pipeline connections to the water pool side, according to the reference passive containment heat removal system. This type of passive system is designed also for the planned Hanhikivi unit in Finland, i.e. for the AES-2006 type nuclear power plant design.

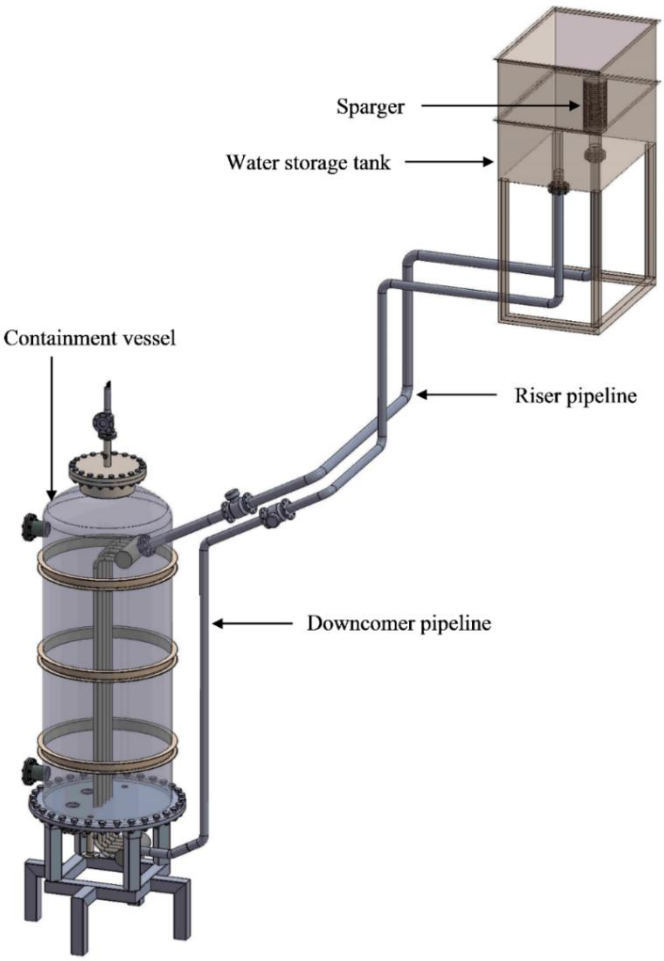


Figure 2.2.5.1. Tentative structure of the LUT test facility.

Deliverables in 2016

- Participating in the OECD/NEA PKL projects
- Research report on the PWR PACTEL test studying effects of nitrogen from an accumulator
- Research report on the simulations of PWR PACTEL nitrogen experiments (published in January 2017)
- Construction plan of the system design to investigate the fundamentals of the selected passive system (published in January 2017)

2.2.6 KATVE - Nuclear criticality and safety analyses preparedness at VTT

The general objective of the KATVE project is 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 are mainly related to reactor physics and radiation transport, but also heat transfer and fuel integrity analyses are included in a comprehensive safety study of a dry storage cask, which will be completed during the four-year project. In practice, the KATVE project involves development and validation of calculation tools required for safety analyses, studying the domestic and international standards and requirements, and performing practical safety analyses which provide valuable experience for the research personnel.

Specific goals in 2016

One of the main objectives in the project is 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 development and testing has been documented in an extensive M.Sc. thesis that was officially accepted in 2016, even though the work was mostly done during the previous year.

The work on photon transport continued in 2016 with the emphasis on variance reduction methods and further validation of the developed functionalities. The variance reduction methods are required to increase the computational efficiency in cases where the particle flux decreases by orders of magnitude between the source and point of interest, which is the intended outcome of a radiation shielding configuration. The methodology is based on weight-windows. At the current stage of development, the weight-windows can be either calculated with an external software and imported to Serpent or produced with a built-in solver utilizing the response-matrix method. The latter option allows the problem to be calculated with Serpent only and is therefore more user-friendly, but it is restricted to relatively simple cases. The work on variance reduction methods in 2016 produced promising yet preliminary results, so the efforts will continue in the future.

In addition, the implementation of a coupled neutron-photon transport mode in Serpent was initiated in 2016. The purpose of the mode is to assess the gamma heating caused by the prompt secondary gamma photons emerging as a consequence of certain neutron interaction reactions. As the first steps, emission of secondary prompt gamma photons as a consequence of neutron interaction reactions was modelled as well as the heat deposition of such photons. These features were tested and found to provide consistent results with respective MCNP6 calculations. Simultaneous neutron and photon transport is not possible yet, however, but the work on more rigorous coupled neutron-photon will continue. Summer students of Aalto University contributed to testing of both the variance reduction methods and the coupled neutron-photon mode.

Two MCNP Photon Benchmark Problems were calculated with Serpent 2 in order to validate its photon transport mode. A Co-60 source skyshine experiment was one of these allowing comparison both against MCNP calculations and experimental data. The agreement between Serpent and MCNP calculations was very good, meanwhile some discrepancy was observed between the calculations and the experimental data. The other modelled experiment consisted of a Cs-137 source and Teflon shields of various thickness. No experimental data was available for this case, but the calculated results were very close to each other. In addition to these calculations, the SINBAD database containing experimental shielding benchmarks was reviewed in order to identify potential cases for further validation efforts. As a result, 21 suitable experiments were discovered.

Another main objective was to analyse the heat transfer in a dry storage cask filled with spent nuclear fuel. The first goal of the analysis was to determine the largest cladding temperature within the storage cask, which was performed in 2015. The next step, performed in 2016, was to determine the temperature distribution at various time points up to 300 years after discharge. The information calculated so far will be used in fuel integrity analyses later in the project. The decay heat source for the heat transfer was obtained through 3D fuel assembly burnup calculation with Serpent. The BEAVRS benchmark was found to provide a suitable model for the calculation. The CFD heat transfer analysis was performed with OpenFOAM for a CASTOR-V/21 dry storage cask, filled with 21 of the spent PWR assemblies. When the peak cladding temperature (PCT) of the stored fuel pins was calculated, the specific question was whether the PCT would remain below the 400°C limit suggested in the U.S.NRC guidelines. According to the calculations, the limit is fulfilled as far as the fuel is not stored into the dry cask earlier than 3.4 years after discharged from the reactor. The Figure 2.2.6.1 illustrates the axial temperature profiles at various time steps, starting from discharge.

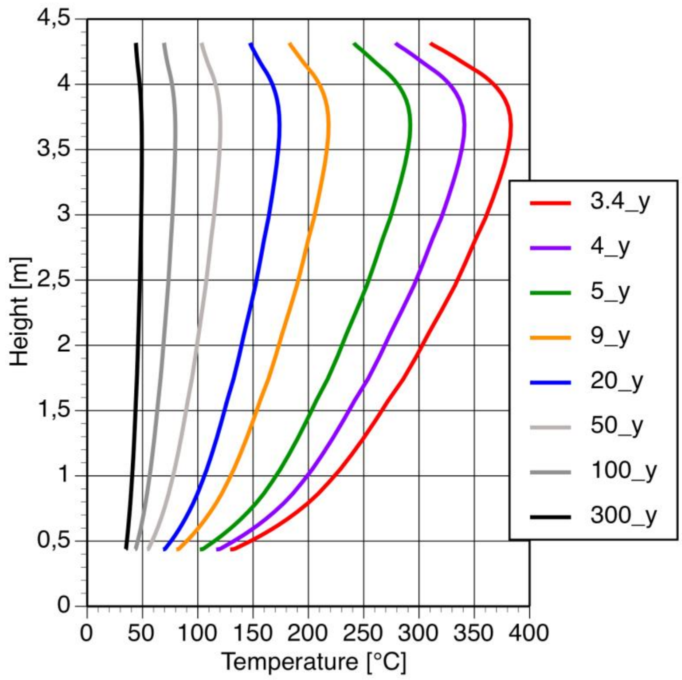


Figure 2.2.6.1: Axial temperature profiles in the most heated region of the studied dry storage cask at various time steps.

Performing valid criticality safety analyses requires that the calculation system, consisting of a transport calculation code and the cross section library, is validated for criticality safety analyses. In practice, this means modelling a large series of criticality 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 increases by the year, and in 2016 the number of available cases was increased to 72 for MCNP and 234 for Serpent. All of the new 10 cases for MCNP were modelled from the same set of experiments. These cases were also added to the Serpent package, in addition to which it was expanded with 34 cases from two series of experiments. Still more independent series need to be added in the collection of experiments before the validation package can be considered complete. Furthermore, the validation script was slightly improved, but further improvements will also be needed.

The codes previously used for activation analyses in the reactor periphery, such as DORT/TORT, are quite cumbersome to use and their support is also about to end, which calls for updating the calculation system. MAVRIC code by Oak Ridge National Laboratories, which combines deterministic solvers together with multi-group Monte Carlo methodology, was recognized as a very potential replacement for the old codes. Preparedness in activation analyses was initiated in 2015 by attending a training course and calculating a simple benchmark study. In 2016, the work was continued by modelling two neutron dosimeter cumulative fluxes within a surveillance chain irradiated in the Loviisa 1 NPP. The calculated activities were compared to previous analyses that have been also verified with measurements. Calculation model contained some simplifications and assumed parameters from public sources, but compared to previous analyses, the results were reasonably close to each other. Results were reported so that the confidentiality of the reactor data was respected, which slightly limited the extent of the reported information.

The same case of surveillance chain activation was also calculated with Serpent 2 and the results were published as a conference paper in proceedings of AER Symposium 2016. The calculated results agreed relatively well with the experimental data. Furthermore, the calculations provided promising results about the efficiency of the variance reduction methods that had been implemented in Serpent 2 recently.

Additionally, related to activation analysis calculation methods studied in KATVE project, an article on previously calculated radionuclide inventories in the FiR-1 research reactor was prepared in 2015 and published in Nuclear Technology journal in April 2016.

One of the goals for year 2015 was to prepare a state-of-the-art report on the burnup credit practices in Finland and abroad. This work was started in 2015, and also a short preliminary version of the report was prepared. The reference group, however, suggested that the report should be completed in its originally intended extent, which was done in 2016, with help from an ad hoc criticality safety group including members from the utilities and the authority. The report covers the basics of the physical phenomena affecting the burnup credit calculations, a summary of the national and the most notable international standards concerning the topic and an overview of the general principles that govern the practical burnup credit analysis.

International collaboration included in the project plan suffered from the bad schedule of the intended meetings, so none of the planned meetings was attended.

Deliverables in 2016

- A conference article accepted in the M&C 2017 conference describing the development of variance reduction methods in Serpent 2.

- A conference article accepted in the M&C 2017 conference about the development of coupled neutron/photon transport mode into Serpent 2 to expand the scope of applicability in radiation shielding analysis.

- A conference article was published in proceedings of the AER Symposium 2016. The article presents the neutron dosimetry calculations, in which Serpent 2 was used to estimate the activation of the surveillance chain irradiated inside the reactor pressure vessel of the Loviisa-1 NPP.

- Report on validating the gamma transport in Serpent. Two test cases were calculated with Serpent: a skyshine experiment and a Cs-137 source with a Teflon shield. For comparison, the same cases were calculated with MCNP6. Suitable cases for further validation calculations were also identified in the report.

- A report on the heat transfer of a CASTOR-V/21 dry storage cask, filled with 21 PWR assemblies at 50 MWd/kgU burnup. The peak cladding temperature (PCT) was calculated as a function of storage time to determine if the remains below the 400°C limit guided in a U.S.NRC document.

- Status report on the development of the criticality safety validation package for Serpent and MCNP in 2016. The package itself can also be considered a deliverable.

- Report on the calculation of surveillance chain activation using MAVRIC code. The calculations were compared to the same experimental data as the Serpent calculations for the AER Symposium paper.

- Report on burnup credit containing, for example, basics of its underlying physics, observations of international standards and regulations as well as practical information about code validation in brief.

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 developed methodologies are thoroughly validated and put to practice in the calculation schemes used at VTT for the independent safety analyses of Finnish power reactors.

Specific research topics include 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. Before moving to these new research topics, the state-of-the-art methodology developed within the KÄÄRME project is finalized and comprehensively validated. In practice this implies moving from proof-of-concept to practical applications. The project shares topics and collaboration with the KATVE, SADE and FURIOUS 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. The plans for MONSOON 2015 were drafted based on these specific recommendations.

Specific goals in 2016

The realized volume of the project in 2015 was reduced to 50% from what was applied, which meant that many of the new research topics and in general the broadening of the range of applications had to be postponed to later years. Instead, the project focused on completing the work started in the KÄÄRME project in SAFIR 2014, which had to be accomplished before moving on to new challenges. The volume for 2016 was reduced even further, to about one third of that requested in 2015, which meant even more reductions in the planned tasks.

The first goal of the project was to develop Serpent into a practical tool for group constant generation, in such way that this task could be handled by relatively inexperienced users with only basic skills in reactor physics. In other words, such that Serpent could replace current deterministic lattice transport codes in the traditional multi-stage reactor physics calculation scheme. In the original plan the work was focused on group constant generation for six simulator codes currently used at VTT:

1. ARES – Steady-state nodal diffusion code for the fuel cycle simulations of western LWR's, developed at STUK
2. TRAB3D – Time-dependent nodal diffusion code for the transient analyses of western LWR's, developed at VTT
3. HEXTRAN – Time-dependent nodal diffusion code for the transient analyses of VVER reactors, developed at VTT
4. HEXBU – Steady-state nodal diffusion code for the fuel cycle simulations of VVER reactors, developed at VTT
5. PARCS – Steady state / time-dependent nodal diffusion code for the fuel cycle and transient analyses of LWR's, developed at Purdue University

The physics of spatial homogenization is very similar for all codes based on nodal diffusion methods, but there are differences in the practical implementation. The cross section model in VTT's simulator codes TRAB3D, HEXTRAN and HEXBU is also used by the Apros system code, so the work also supports the calculation scheme for nuclear power plant simulations.

Because of the reduction in project volume, the research topics had to be limited to work already started in SAFIR 2014, involving the first two simulator codes in the list - ARES and TRAB3D. Work on the Serpent-ARES calculation sequence was started during the last year of the KÄÄRME project, with the MIT BEAVRS benchmark involving a 1000 MW Westinghouse PWR as the test case. In MONSOON 2015-2016 the calculations were extended from hot zero-power initial core calculations to full power conditions and fuel cycle simulations. The calculations were completed in early 2016, and the results were reported in a paper published in Annals of Nuclear Energy.

The Serpent-TRAB3D calculation sequence was demonstrated earlier using the initial core of the EPR reactor as the test case. The plan for 2015 involved validation of the pin-power reconstruction module in TRAB3D, followed by studies on the effects of partially-inserted control rods and other axial heterogeneities in the accuracy of nodal diffusion calculations. The work was co-funded with the SADE project and carried out as a special assignment and an M. Sc. thesis for Aalto University. The thesis was completed in 2016.

Work carried out in collaboration with the SADE project also involved modeling the zero-power critical facility V-1000 at Kurchatov Institute using HEXTRAN and Serpent 2. The results were presented at the AER 2016 symposium.

Additional funding was received from Fennovoima in late 2016 for testing the Serpent-HEXBU calculation sequence. The topic was included in the original project plan, but had to be dropped because of considerable budget cuts. The test case for the calculation was an initial HZP core of a VVER-1000 reactor (Coolant Transient Benchmark of Kozloduy Nuclear Power Plant Unit 6). The work was documented in a VTT project report.

Including fuel temperature feedback in the assembly burnup calculation performed for the purpose of spatial homogenization was listed in the original work plan as one of the major research topics. Because of the reduction in budged, the work was focused on fuel performance code coupling, leaving studies related to new approaches to state-point parametrization to later years. The work began in 2016 with the external coupling of Serpent to the ENIGMA fuel performance code. The coupled depletion methodology was first tested with a Peach Bottom 2 assembly and later by generating group constants for an EPR full core ARES-model with Serpent using accurate fuel temperature distributions. The effect of the accurate fuel temperature distributions on nuclide inventories, generated group constants and finally the results of the nodal code (ARES) was estimated by comparing the results to simulations conducted with the traditional flat, constant fuel temperature history. The work was reported in two papers submitted to Annals of Nuclear Energy and the M&C2017 conference.

In addition to the actual research topics, work was also allocated to dealing with practical challenges in spatial homogenization, in particular the development of an automated burnup sequence capable of handling state-point variations. The process of group constant generation requires repeating the assembly-level transport calculation hundreds or even thousands of times, in order to cover all fuel assembly types and reactor operating conditions. The automated management of input and output data is an absolute necessity for accomplishing this task.

The project also included sub-tasks for international collaboration and the preparation of Serpent 2 for public release. These sub-tasks involved participation in the activities of the international reactor physics community, organization of Serpent user group meetings, as well as the preparation of an input manual and new cross section libraries for Serpent 2.

Deliverables in 2016

All tasks planned for 2016 were completed. Specific deliverables are listed below.

- The Serpent-ARES calculations involving the MIT BEAVRS Benchmark were extended to hot full-power conditions and fuel cycle simulations. The comparison of ARES results to reference Serpent 3D calculations showed good agreement for radial and axial power distributions for the HFP state. Control rod worths and boron dilution curve for the first operating cycle were in good agreement with experimental measurements provided in the benchmark specification. Most of the calculations were completed in 2015 and reported in a paper published in Annals of Nuclear Energy in 2016.

- The ARES calculations demonstrated that it is possible to use the continuous-energy Monte Carlo method for producing the full set group constants for fuel cycle simulations, albeit at a high computational cost. It was also demonstrated that the automated burnup sequence developed during the course of the project for this purpose works as intended. The burnup sequence, and the methodology used in Serpent for group constant generation, were covered in a paper published in Annals of Nuclear energy in 2016.

- Work on assembly burnup calculations with fuel temperature feedback was started by coupling Serpent to the ENIGMA fuel performance code. The calculations were reported on two papers submitted to Annals of Nuclear Energy and the M&C 2017 conference.

- An M. Sc. Thesis for Aalto University on the modeling of axially heterogeneous systems using nodal diffusion methods (TRAB3D) was completed in 2016. The work was co-funded with the SADE project.

- The international Serpent user community grew from 500 users in January 2016 to 600 users by the end of January 2017. The code has users in 175 universities and research organizations in 37 countries around the world. The Serpent website lists more than 400 peer-reviewed scientific journal articles and conference papers and 120 theses published on Serpent-related topics worldwide.

- Two source code updates (2.1.26 and 2.1.27) were distributed to Serpent during the course of the project.

- The 6th Annual Serpent User Group Meeting was hosted by the Politecnico di Milano in Milan, Italy, on September 26-29, 2016. The meeting brought together more than 40 Serpent users from 24 Organizations around the world.

- A Serpent workshop was organized at the PHYSOR 2016 conference in Sun Valley, Idaho, USA, in May 1-5, 2016.
- A Serpent workshop was organized at the AER 2016 symposium in Helsinki, in October 10-14, 2016.
- International collaboration also included participation in the Executive Committee of the Reactor Physics Division of American Nuclear Society. Two meetings were attended in 2016.
- A Ph.D. student from Institut Jozef Stefan (JSI) in Slovenia visited VTT in September 2016. The purpose of the visit was to exchange information on spatial homogenization and familiarize JSI in the use of the Serpent-based calculation sequence
- Writing of a User Manual for Serpent 2 as an on-line Wiki was continued in 2016.
- Several papers from Serpent user organizations were co-authored in 2016.
- Ville Valtavirta's doctoral thesis “Development and applications of multi-physics capabilities in a continuous energy Monte Carlo neutron transport code” was submitted for pre-examination. Thesis defense will take place in May.

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

In the NURESA project, Computational Fluid Dynamics (CFD) methods are developed and validated for the identified most important topics in nuclear reactor safety assessment. In Work Package 1 (WP1), international single-phase mixing and stratification benchmarks are participated. In WP2, PPOOLEX spray and stratification experiments are modelled with CFD codes in co-operation with LUT and Swedish partners. In WP3, CFD models for departure from nucleate boiling (DNB) and dry-out are developed for the OpenFOAM code in co-operation with international partners. In WP4, two-way coupled CFD-Apros calculations of nuclear power plant steam generator are performed. WP5 consists of the coordination of the project.

Specific goals in 2016

In WP1, the international blind benchmark in the HYdrogen Mitigation Experiments for REactor Safety (HYMERES) programme was participated in 2015. The blind benchmark exercise was PANDA experiment HP1_6_2, which focussed on the hydrogen stratification and erosion of density layer by turbulent mixing processes. The benchmark results are published in 2016.

In WP2, thermal stratification experiments performed at LUT with the PPOOLEX facility are modelled with CFD simulations. Experiment with safety relief valve (SRV) sparger is studied, where the thermal stratification and mixing of the pressure suppression pool during steam injection into the pool is calculated. The numerical results are compared to the PPOOLEX experiment.

In WP3, OpenFOAM CFD solver is developed and validated for nuclear reactor safety assessment. At VTT, boiling models are developed and integrated to the Eulerian two-phase solvers of the official OpenFOAM release. At Aalto, heat transfer in fuel rod bundles is calculated with OpenFOAM. At LUT, OpenFOAM simulations of POOLEX chugging tests are done and models for direct-contact condensation are developed. In the in-kind contribution of Fortum, heat transfer in VVER-440 fuel rod bundle is calculated.

In WP4, coupled CFD-Apros simulation of a steam generator is performed. In the SGEN project of the SAFIR2014 programme, models of steam generators have been developed, where the primary circuit has been modelled with Apros and the secondary side of the steam generator has been modelled with ANSYS Fluent CFD code. In the previous calculations, one-way coupling of Apros and Fluent was used, which in the simulations of long transients lead to discrepancies between the Apros and CFD models. Therefore, two-way coupling of the codes is needed. Transient in a VVER-440 steam generator is calculated as the test case.

In WP5, project coordination is done. In addition, Northnet Roadmap 1 and Roadmap 3 Reference Group Meetings are participated. Journal article on the steam generator modelling performed earlier in SAFIR2014 programme is written.

Deliverables in 2016

- Synthesis of international blind CFD benchmark exercise was published in CFD for Nuclear Reactor Safety Applications, CFD4NRS-6 Workshop. The benchmark was based on a test in the PANDA facility addressing the stratification erosion by a vertical jet in presence of a flow obstruction. The exercise was participated by the NURESA project in 2015.

- The thermal stratification of the pressure suppression pool during steam injection into the pool was studied with CFD calculations. Stratification and mixing experiment SPA-T1 performed at LUT with PPOOLEX facility was calculated. In the experiment, steam was injected into the water pool through a sparger. In the beginning, steam was injected into the pool with a small mass flow rate of 30 g/s, which caused stratification of the pool (see Figure 2.2.8.1). After the stratification period of about three hours, the mass flow rate of steam injection was increased to 123 g/s, which caused mixing of the pool. Results of the CFD calculation were in fairly good agreement with the experiment.

- At VTT, subcooled nucleate boiling and wall heat transfer models have earlier been integrated to the Eulerian two-phase solvers of the official OpenFOAM release. The boiling models have now been extended to higher void fractions, where the longer-term goal is modeling of Departure from Nucleate Boiling (DNB). Calculation of DNB situation is illustrated in Figure 2.2.8.2. Interfacial Area Transport Equation (IATE) models have been tested in boiling simulations and found to produce promising results at high void fractions. Robustness of the solver in thermal phase change simulations has been significantly improved. Reports have been written documenting the implementation and the validation of the boiling solver that will be included in the official OpenFOAM release.

- At LUT, two-dimensional axisymmetric CFD simulations of PPOOLEX DCC-05 experiment have been performed with compressible two-phase OpenFOAM CFD solver. Various modelling issues including the performance of interfacial heat transfer models, the influence of turbulence modelling, interfacial momentum transfer, geometry and interface initialization have been studied. Turbulence modelling in OpenFOAM calculations is illustrated in Figure 2.2.8.3. The OpenFOAM results have been compared to the test results and corresponding NEPTUNE_CFD simulations. The bubble volume and the chugging frequency during the blowdown were obtained from the test data by using the pattern recognition algorithm developed at LUT. Report on the DCC simulations of PPOOLEX experiments has been written.

- At Aalto University, new coarse grid was made for the VVER-440 fuel-rod bundle using symmetry boundary conditions and a full-length bundle. Single-phase simulations of the fuel-rod bundle were performed with different turbulence models. The heat balances and sub-channel temperatures in the simulations were studied. The temperature distributions of single-phase calculations gave an indication, where sub-cooled nucleate boiling starts. Two-phase simulations were also made for VVER-440 fuel rod bundle. Report on the simulation results was written.

- At Fortum, heat transfer calculations for VVER-440 fuel rod bundle have been performed. This part of the work is an in-kind contribution of Fortum to the project. First, the full fuel rod bundle model was updated and ANSYS Fluent simulations were made for the verification of the model. Second, OpenFOAM calculations for full bundle were performed. Report on the simulation results was written.

- CFD-Apros simulation of a steam generator transient has been performed by using two-way coupling of the codes. Main Steam Line Break (MSLB) has been calculated by coupling generic Apros model of VVER-440 plant with CFD model of the secondary side of the steam generator. The couplings of the CFD model with the plant model are the following: (i) heat transfer from the primary tubes of the Apros model to the CFD model, (ii) feed water injection from the Apros model to the CFD model, (iii) surface level measurement based on pressure difference in the CFD model, (iv) two-phase flow from the steam generator to the steam line. The simulation provided detailed information on the behavior of the steam generator after the main steam line break, which was followed by turbine trip and reactor scram. The steam generator model is illustrated in Figure 2.2.8.4.

- Project was coordinated by VTT. Northnet Roadmap 1 meetings were participated in June and December. Northnet Roadmap 3 meeting was participated in March. Journal article on the steam generator modelling performed earlier in SAFIR2014 programme was written.

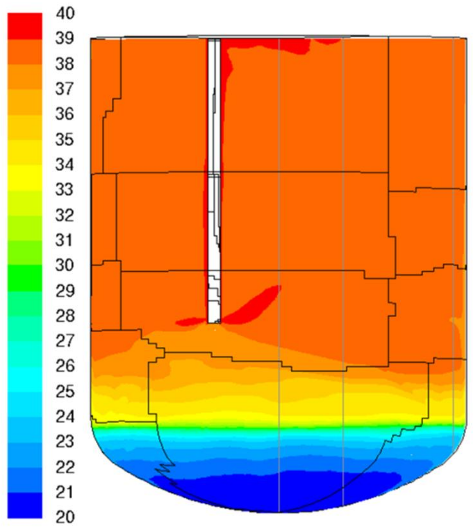


Figure 2.2.8.1. Temperature (℃) of water in thermally stratified pressure suppression pool of the PPOOLEX facility.

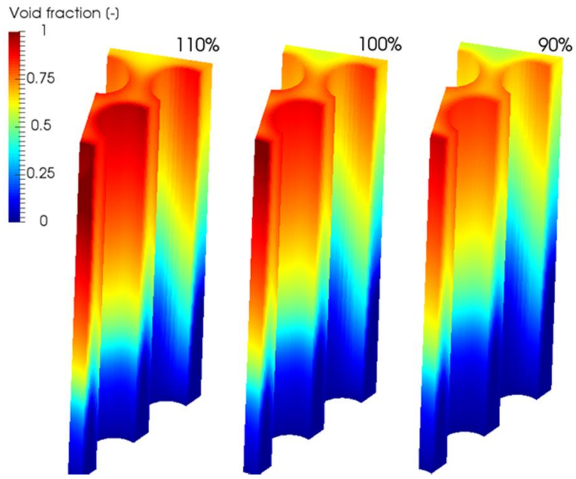


Figure 2.2.8.2. Void fraction during boiling in the fuel rod bundle of Large Water Loop experiment. Results at three different power levels are shown, where 100% corresponds to DNB.

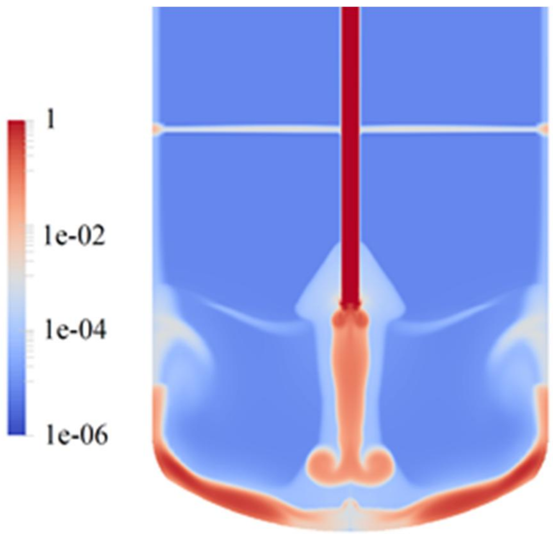


Figure 2.2.8.3. Turbulence kinetic energy (m^2/s^2), when steam flows into water pool through a vertical vent pipe. OpenFOAM calculation of a PPOOLEX experiment.

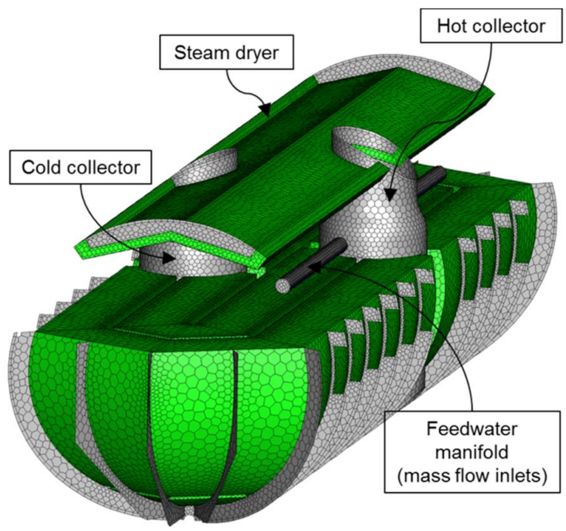


Figure 2.2.8.4. CFD model of the internals of VVER-440 steam generator. The primary tubes and the steam dryer are described as porous media which is shown in green color.

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 UO2 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 2016

Focus for 2016 was in development of FINIX fuel code and the validation system, the verification of the coupling of SCANAIR to thermal-hydraulic sub-channel code GENFLO and the investigation of cladding properties with the CREBELLO device. The international co-operation 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 progress is done under this project.

Deliverables in 2016

- Development of steady state heat transfer models to FINIX. The steady state fuel behaviour solution is naturally required if the coupled problem being solved is a time-independent one, but it also provides the initial conditions for transient simulations. As the use of a time-dependent solver with a very large timestep to obtain the steady state solution was seen to be both computationally wasteful and a possible source for convergence difficulties a separate steady state solver was implemented.

  The new steady state solver used a similar linearization of the material heat conductivities and power density as the time-dependent solver in order to produce a temperature solution very close to the one given by the transient solver. The implementation of the steady state solver was successful, with verification conducted against the transient solver.

- New fission gas release model has been implemented to FRAPCON 4.0. The model is based on the work of Giovanni Pastore and is similar to his model in BISON code.

- In 2016 the SPACE validation tool was completely rewritten based on the software development plan that was created in 2015. The new validation tool is composed of a relational database and a piece of software for simulation program validation. The redesigned database contains all the data that are needed to construct the simulation program input files. It also contains the experimental data to which the simulation results can be compared.

- Fuel Modelling in Accident Conditions (FUMAC) is an IAEA Coordinated Research Project (CRP) that was launched in 2014. The aim of the project is to support the participants from different countries in their efforts to develop reliable tools for modelling of fuel behaviour during accidents. VTT’s contribution in the first phase of the project was to calculate a set of AEKI separate effect tests and an integral VVER LOCA test performed at the Halden reactor. The code used in the calculations was FRAPTRAN-GENFLO. The preliminary results calculated in the first phase of the FUMAC project guided the discussion in the second research coordination meeting that was held in May 2016. The meeting allowed the participants to share in-formation and to plan the final phase of the project. The data, information and expertise that will be gained during the FUMAC project will be essential in developing and validating the VTT's in-house fuel behaviour module FINIX.

- The coupling between SCANAIR and GENFLO was introduced in 2014 in PALAMA project as a yearly in-kind work for IRSN. The coupling was successfully implemented in a way that GENFLO calculates the cladding outer surface temperature in addition to the thermal hydraulic behaviour, and passes the axial cladding outer surface temperatures on-line to SCANAIR. Then, SCANAIR simulates the fuel thermo-mechanical + FGR behaviour. With the thermal-hydraulic coupling, improvement to the existing SCANAIR modelling is anyhow evident. In 2016 the coupling has been improved and several issues resolved. A journal manuscript has been prepared on SCANAIR-GENFLO coupling, complemented with updated results on BWR cladding low temperature failure analysis made in 2012.

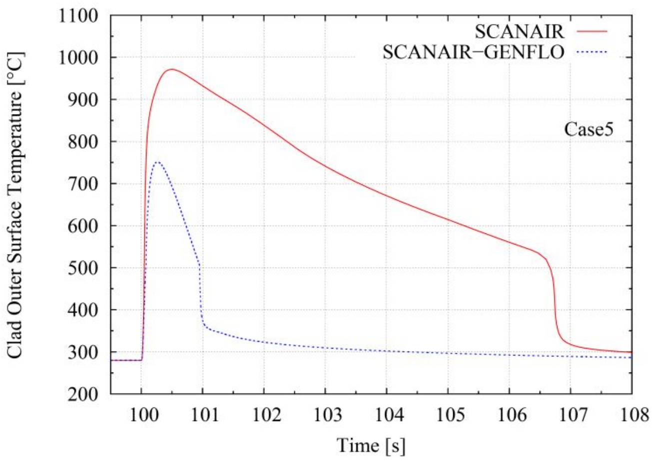


Figure 2.2.9.1. Cladding outer surface temperature in SCANAIR-GENFLO vs. stand-alone SCANAIR in RIA benchmark PWR case with boiling.

- Phase II of the RIA modelling codes benchmark is divided into two Activities. In 1st Activity, 10 simplified RIA cases were simulated. In 2nd Activity, uncertainty and sensitivity analysis was done for a selected case from Activity 1. The number of uncertainty runs was 200. VTT participated to both activities with SCANAIR V_7_4 and V_7_5 by simulating all the specified simplified cases and uncertainty runs, and by performing the sensitivity analysis.

  Phase II of the benchmark was initiated in spring 2014 followed by the first meeting in September 2014 (Paris) and the second in April 2015 (Brussels). Third meeting of Phase II was held in September 2015 in Paris. The final meeting to present and discuss the final results and the draft report was held on 20-21 June 2016 in Lucca, Italy.

- The previous experiments with ThO2 pellets were presented in a journal article as well in the 9th international conference on nuclear and radiochemistry.

- A creep strain modelling approach (Logistic Creep Strain Prediction LCSP) for nuclear fuel cladding material Zircaloy-4 was reviewed and evaluated in steady state and transient creep conditions. These results were presented at Baltica X conference.

- The second creep experiment utilizing 1%Nb Zirconium alloy tube with alternating stresses was performed with CREBELLO device. This experiment has been reported.

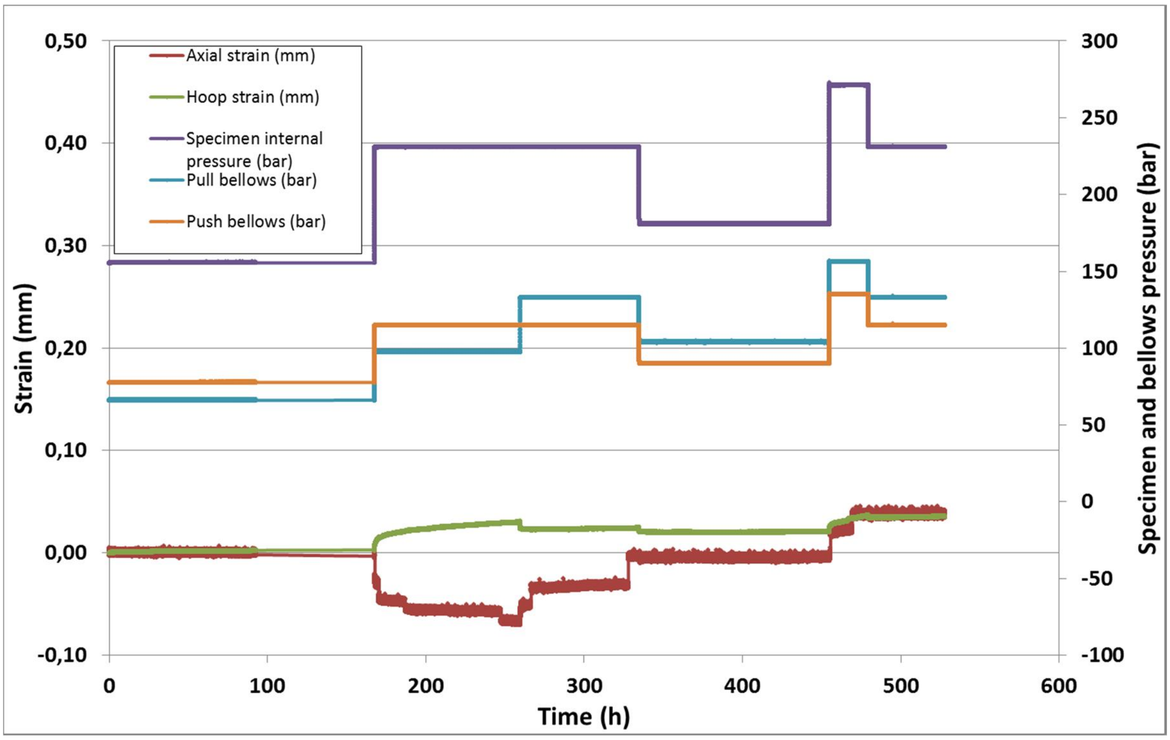


Figure 2.2.9.2. The curves for axial strain, hoop strain, specimen internal pressure, pull bellows and push bellows in a transient creep test.

- HPG and WGFS meetings were attended and reported to the reference group.

In addition to this, a half day seminar was held on August 10th discussing national and international projects and progress in the field of nuclear fuel research.

2.2.10 SADE - Safety analyses for dynamical events

Aim of SADE is 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 will be improved by routine 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 a 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 will be tested and demonstrated in cases relevant from safety analyses point of view. Objective is 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 2016

The project has two main research areas. The objective of the first work package is to enhance the neutronics modelling of VTT's 3D reactor dynamics codes. Development of the solution methods and analysis of the methods versus accurate reference results is also an efficient way to study nodal codes in depth. In addition, the aim is also to have at VTT a fully self-developed, independent calculation system which can be used for the whole calculation sequence from basic nuclear data to coupled 3D transient analyses. During the SAFIR2018 the project aim is that group constants can be routinely created with the reactor physics code Serpent 2 for the reactor dynamics codes.

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. Tools that enable more realistic modelling of the transients will be further developed and transients will be simulated with these improved tools. Modelling and development has two parallel branches: development of the tools such as internally coupled HEXTRAN-SMABRE that could 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 2016, two-way coupling of CFD-style calculation to existing calculation tools will be demonstrated.

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 2016

- Master’s thesis “Modelling of axial discontinuities in reactor cores with Serpent 2 – TRAB3D code sequence” has been completed. The differences between TRAB3D and Serpent 2 have been reduced by 58.9 % in hot zero power EPR core case. There are reasons to assume that to further improve the accuracy of TRAB3D, the neutronics model would have to be changed.
- Conference paper on Serpent - HEXTRAN code sequence was presented at AER 2016 Symposium in Helsinki, Finland. The studied case was the zero-power critical facility V-1000 in Kurchatov Insitute. The conference paper was accepted for publication in the KERNTECHNIK journal in the KERNTECHNIK AER Issue in August 2017.
- Two CFD grids for VVER-1000 reactor pressure vessel were created. The coarse mesh consists of approximately 280 000 cells and the less coarse mesh consists of approximately 570 000 cells.
- Routines for one- and two-way data exchange between PORFLO and HEXTRAN were developed. The coupled codes were tested using VVER-1000 Coolant Transient (V1000CT) benchmark.
- SMABRE-PORFLO and HEXTRAN-PORFLO couplings were developed further with very limited funding and test simulations were continued using V1000CT and AER benchmark 7 (AER-BM7).
- Coupled HEXTRAN-SMABRE-PORFLO models and the coupled cimulations were presented as conference paper and presentation in the AER 2016 Symposium in Helsinki, Finland. The conference paper was accepted for publication in the KERNTECHNIK journal in the KERNTECHNIK AER Issue in August 2017.
- The project included participation in the AER working group D meeting, where preliminary results of the HEXTRAN-SMABRE-PORFLO calculations were presented.

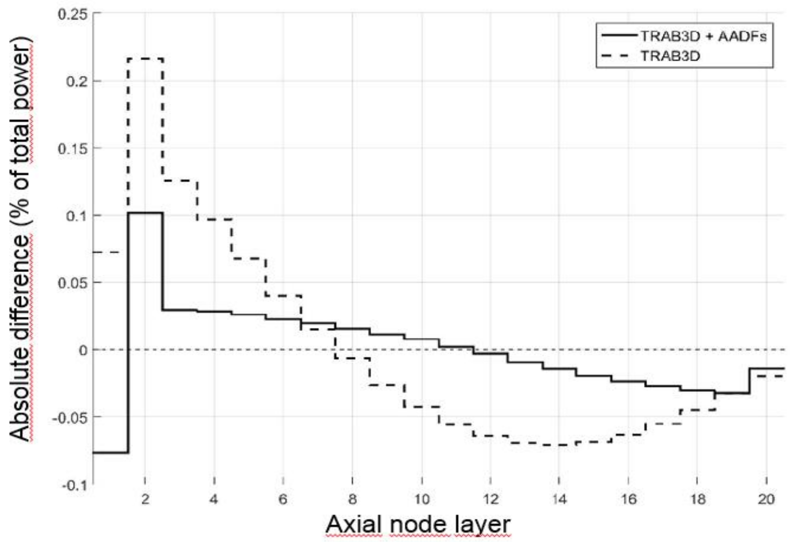


Figure 2.2.10.1: The absolute differences in the axial power distributions of TRAB3D and Serpent 2 for two TRAB3D calculations of hot zero power EPR core, with and without axial discontinuity factors. The peak difference between TRAB3D and Serpent 2 was reduced by 52.9 % with the use of axial discontinuity factors.

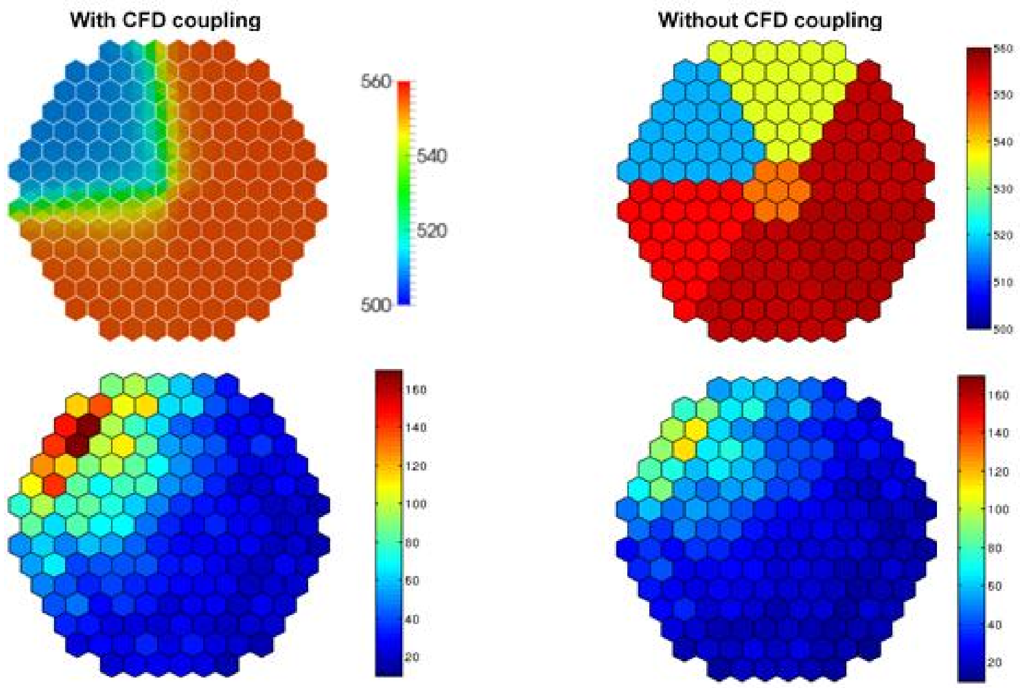


Figure 2.2.10.2: Core inlet temperature (top) and assembly-wise fission power (bottom) in the CFD coupled HEXTRAN-SMABRE-PORFLO (left) and HEXTRAN-SMABRE (right) simulations for a VVER-1000 transient (OECD/NEA Benchmark V1000CT-2 Exercise 3). With CFD, the maximum assembly-wise power peaks roughly 30% higher than with traditional approach.

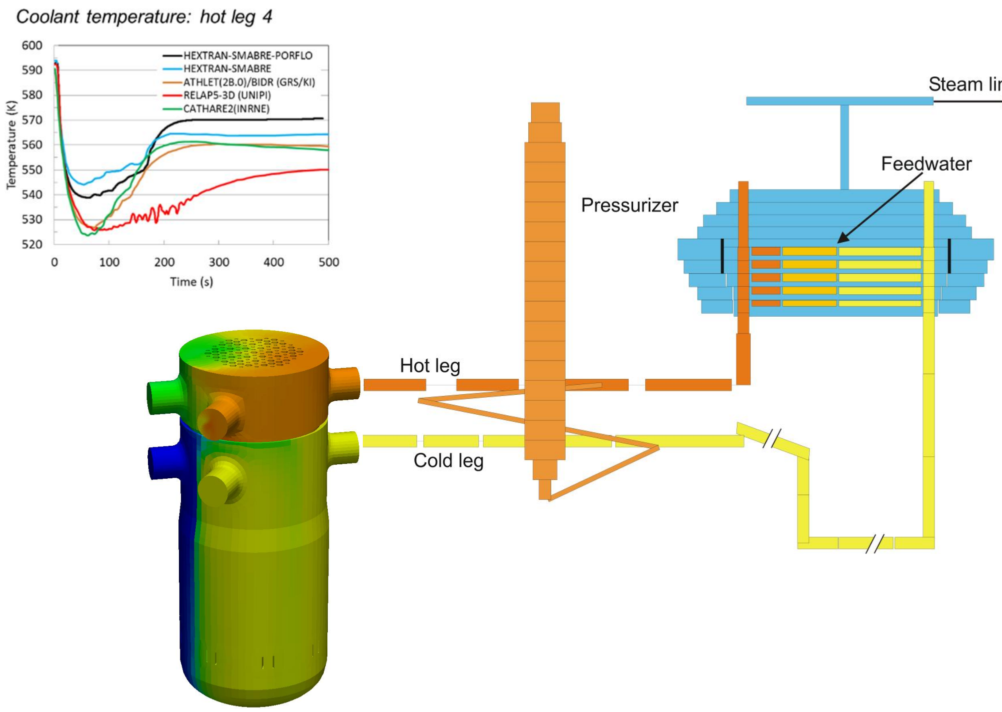


Figure 2.2.10.3: Schematic representation of coupling the CFD modelling domain for the pressure vessel (coloured by the coolant temperature) with the SMABRE nodalization (nodes shown only for one loop) in a coupled simulation of a VVER-1000 plant. Top left corner: Coolant temperature in the hot leg 4 in a fully coupled HEXTRAN-SMABRE-PORFLO simulation for a VVER-1000 transient with the results of other codes (OECD/NEA benchmark V1000CT-2 - Exercise 3).

2.2.11 USVA - Uncertainty and sensitivity analyses for reactor safety

The general goal of the USVA project is to develop methods and practices in uncertainty and sensitivity analyses of multi-physics 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 builds on the existing expertise in uncertainty and sensitivity analysis at VTT and Aalto University, and gathers the on-going research activities under one project. Also new experts in this area are trained. USVA promotes activities at the interfaces of the different disciplines in reactor safety.

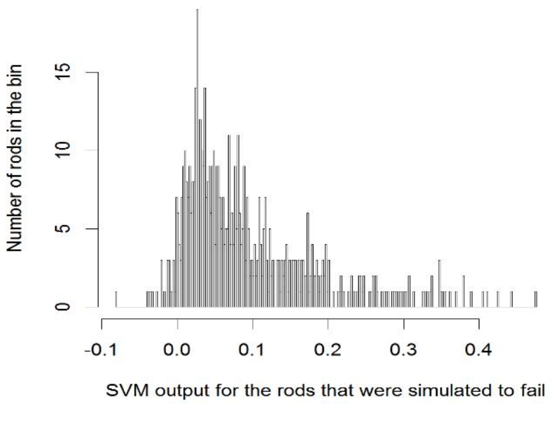
In 2016, two scientific papers were prepared: conference paper on statistical and sensitivity analysis on LOCA, and a journal manuscript on CASMO-4 – SIMULATE-3 uncertainty propagation sequence. Also, a Master’s thesis was finalized at Aalto University. In addition, two extensive literature reviews on state-of-the-art methodology for uncertainty analysis for fuel behaviour and thermal hydraulics were conducted.

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

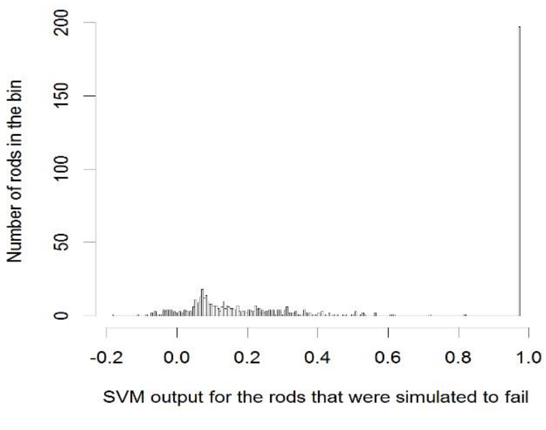
Specific goals in 2016

The LB-LOCA scenario in an EPR type power plant has been previously evaluated at VTT with statistical methods. In the analysis, the FRAPTRAN-GENFLO fuel behaviour and thermal hydraulics code was used to estimate the percentage of failing rods in 59 global scenarios. Each of the global scenarios involved calculating boundary conditions for the transient with APROS. In each of the global scenarios, 1000 FRAPTRAN-GENFLO simulations were performed, and in the worst scenario with respect to the number of failing rods, all the 63 835 rods were simulated. This data was made use of in the sensitivity analysis in USVA.

In 2016, a comprehensive literature review on internationally applied uncertainty and sensitivity analysis methods in fuel modelling was done. The possibility of using machine learning via applying support vector machines (SVMs) to the fuel performance modelling in the statistical analysis methodology was studied and an analysis methodology was developed. The analysis methodology utilising SVM's is divided into four phases: in the first phase, an SVM is fitted using the existing data from the 1000 FRAPTRAN-GENFLO simulations. The sensitivity analysis conducted in 2015, helps in selecting the relevant input variables used for fitting the SVM. In the second phase, the fitted SVM is applied with new input parameter values to obtain more predictions of rods that could possibly fail. In the third phase, additional FRAPTRAN-GENFLO simulations are made using the input values of those rods that the SVM predicts to fail. The third phase produces more simulation results regarding the failing rods, and the SVM may be improved with this data. Fitting the SVM anew is the last phase. With this procedure, and by using SVMs, the accuracy of the predictions is improved substantially, as seen in Fig. 2.2.11.1. In the coming years of USVA, the SVM's will be applied for each global scenario to capture the global effects. In addition, the SVM model can be used in sensitivity analysis within the global scenarios.



(a)



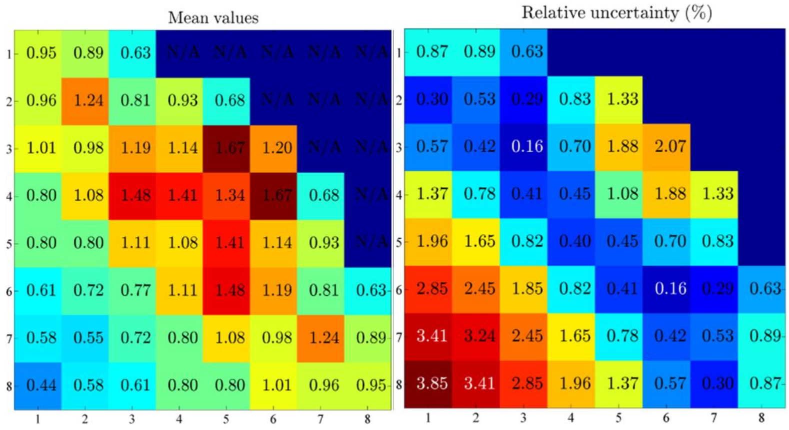
(b)

Figure 2.2.11.1. SVM outputs that correspond to failed rods in FRAPTRAN-GENFLO simulations of all those rods in the reactor that had been in-core for two cycles. The first SVM predictions (a) did not succeed so well (zero stands for survival of rod, unity being the rod failure), but after selecting new cases with the SVM that are susceptible to failure (SVM predictions > 0.1) and training the SVM anew, the SVM predictions are substantially improved (b). Here, SVM was used in regression mode, which produces non-binary outputs. In classification mode, binary information with probabilities whether the rod is predicted to fail or not, is obtained.

In the OECD/NEA PREMIUM benchmark, it was distinguished that the methods for the uncertainty assessment of thermal hydraulics codes' models needs to be further studied. Particularly, the identification of the uncertain input parameters and defining their PDFs for the physical models that the thermal hydraulic safety codes rely on, is challenging. These models have been built based on fitting to experimental data and laws of physics. To start with, a literature review was prepared in 2016. In the review, the existing methods used to quantify the uncertainty in thermal hydraulics codes were studied. In addition, methods used in other disciplines were considered and their applicability to nuclear safety codes were studied. The first objective of the review was to be able to identify the most promising approaches and to set up guidelines on the research for the upcoming years. In the report, the methods introduced in the PREMIUM benchmark were briefly presented before standard methods for solving inverse problems in the field of statistical inversion theory were described. A combination of existing methods in engineering physics is proposed as the solution to the quantification problem. The work will continue based on the recommendations of the review.

In the framework of coupled code systems, the objective was to study the combined uncertainty analysis of coupled neutron transport and fuel behaviour codes for a simple test case. A Master's thesis was written on the subject. Firstly, a computational system coupling the fuel behaviour code FINIX and the reactor physics code DRAGON was set up and applied to the PWR pincell test case of UAM benchmark. Secondly, the nuclear data code NJOY was coupled to the calculation system to allow uncertainty analysis. The most important aspect was to find out if the uncertainties in nuclear fuel modelling and neutronics modelling may be handled separately. As an outcome, these may indeed be propagated separately. A methodology named "CFENSS-SRS" (Coupled Fuel Behaviour and Neutronics Stochastic Sampling with Simple Random Sampling) was developed for the purpose of this work. The method applies the statistical uncertainty analysis to univariate nuclear fuel parameters and correlated neutron cross sections.

In the context of calculation sequences, the idea was to continue the work carried out in the CRISTAL project of SAFIR 2014 programme. Adjoint-based sensitivity and uncertainty analysis capability had previously been implemented to the assembly-level reactor physics code CASMO-4. In USVA, the implementation enabled the uncertainty analysis of assembly constants that were then passed on to codes simulating a full reactor core. In order to be able to propagate uncertainty through core-level simulations in a consistent manner, the methodology was extended to reflector regions. An automated calculation system was developed for propagating nuclear data uncertainty through assembly-level homogenization calculations with CASMO-4 in fresh fuel cases. Nuclear data uncertainty was propagated through the CASMO-4 – SIMULATE-3 calculation chain (see Fig. 2.2.11.2). A journal article on the CASMO-4 -- SIMULATE-3 uncertainty propagation sequence was finalized and the manuscript was submitted to Annals of Nuclear Energy.



(a) (b)

Figure 2.2.11.2. (a) Radial power distribution, and (b) respective uncertainties for a PWR modelled using CASMO-4 -- SIMULATE 3 uncertainty analysis sequence.

Deliverables in 2016

- A conference article (*Statistical and sensitivity analysis of failing rods in EPR LB-LOCA*) and presentation in the TopFuel 2016 conference concerning the statistical and sensitivity analysis of failing rods in EPR LB-LOCA.

- A journal article (*Sensitivity analysis of local uncertainties in large break loss-of-coolant accident (LB-LOCA) thermo-mechanical simulations*) was revised based on reviewer comments and re-submitted to Nuclear Engineering and Design.

- A research report on the application of Support Vector Machines as a surrogate model to fuel performance simulations to predict fuel failures in LOCA.

- A literature review on state-of-the-art uncertainty and sensitivity analysis methods in nuclear fuel modelling.

- A literature review on state-of-the-art methods for uncertainty quantification regarding input parameters of physical models in thermal hydraulic codes.

- A Master’s thesis was completed on the subject of the combined uncertainty analysis of coupled neutron transport and fuel behaviour codes utilizing a novel CFENSS-SRS (Coupled Fuel Behaviour and Neutronics Stochastic Sampling with Simple Random Sampling) methodology.

- A journal article was submitted to Annals of Nuclear Energy describing the uncertainty propagation in the CASMO-4E/SIMULATE-3 code chain with the title “*Uncertainty analysis of assembly and core-level calculations with application to CASMO-4E and SIMULATE-3*”.

- Travel report on the UAM-10 benchmark workshop.

2.3 Structural safety and materials

In 2016 the research area "Structural safety and materials" consisted of eight 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)

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

The general objective of the ERNEST project is to enhance domestic capabilities for external event assessment through computational and material model updating and validation with the aid of experimental test data, gathered also within the project. The external event mainly considered in the project is an aircraft crash against nuclear power plant structures. These impacts pose numerous threats for the safety of the plants. The computational models that are used are mainly finite element models with which these crashes can be studied in detail and their effect on the safety of the plant assessed. In order to be reliable, these models need to be validated against relevant and reliable data from experimental tests. In order to be of help in validation, the experimental tests have to evoke the same phenomena than the real aircraft crash does, though all the phenomena need not be present in every test.

Specific goals in 2016

One of the main goals in 2016 was to carry out two tests with reinforced concrete slabs and gather valuable experimental data for validation of computational models. The tests were carried out by impacting a semi-soft projectile against a reinforced concrete wall. The response of the slabs was measured with strains gauges glued on the reinforcement bars as well as on the front surface of the slabs and with displacement sensors. The behaviour of the slabs combined global bending and local shear punching. In the first test, the emphasis was more on the bending response while the second test evoked remarkable shear punching response. The tests yielded wealth of useful data for computational model validation. They also illustrated the difficulty of assessment of shear punching damage.

Another main goal in 2016 was to enhance the capabilities of Abaqus finite element code for modelling concrete material. There are several reasons why accurate modelling of concrete material is important for aircraft impact simulations. These include the following:

- Concrete is a difficult material to model (heterogeneous, anisotropic, porous, multi-phase, multi-scale, brittle, …).

- It's behavior under impact is poorly known (strain rate effect, inertia effect, capillary effect, high speed crack propagation, …).

- Effect of concrete material model parameters is predominant in large scale structural impact simulations.

What was carried out in 2016 was as follows:

- A literature survey of concrete modelling was executed including
    - thermodynamic damage-plasticity continuum approach,
    - the ”Barcelona” concrete model (J Lubliner, et Al. A plastic-damage model for concrete. Int.J. of Solids and Structure, 1989.) and
    - the ”Lee-Fenves” concrete model (J.H. Lee and G. Fenves. Plastic-damage model for cyclic loading of concrete structures. Journal of Engineering mechanics, 1998.).
- Enhancement of the Abaqus CDP model including
    - a confinement stress dependent concrete model (T. Gabet. Thèse: Comportement triaxial du béton sous fortes contraintes: influence du trajet de chargement. Université Joseph Fourier, Grenoble, 2006.) and
    - study of element deletion as a way of materializing cracks in impact simulations.

Deliverables in 2016

- A research report on the concrete material modelling.
- A research report on the experimental impact tests.

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: 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 are the direct applications to NPPs. The results of FIRED work packages are illustrated in Figure 2.3.2.1.

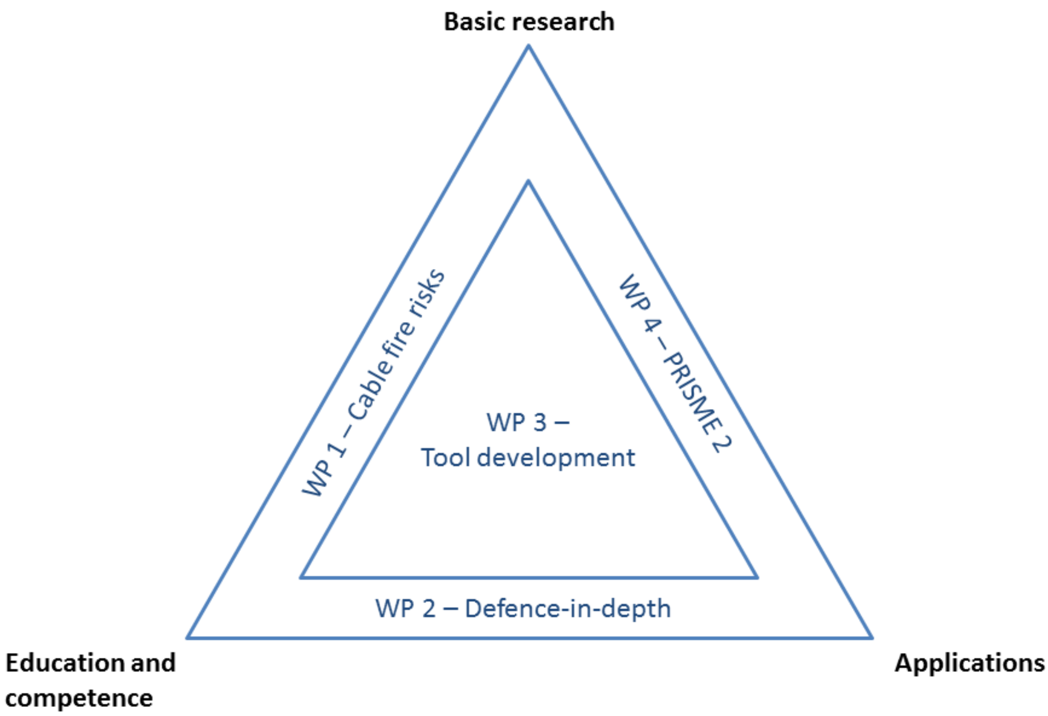


Figure 2.3.2.1. Result categories in WPs of FIRED.

Specific goals in 2016

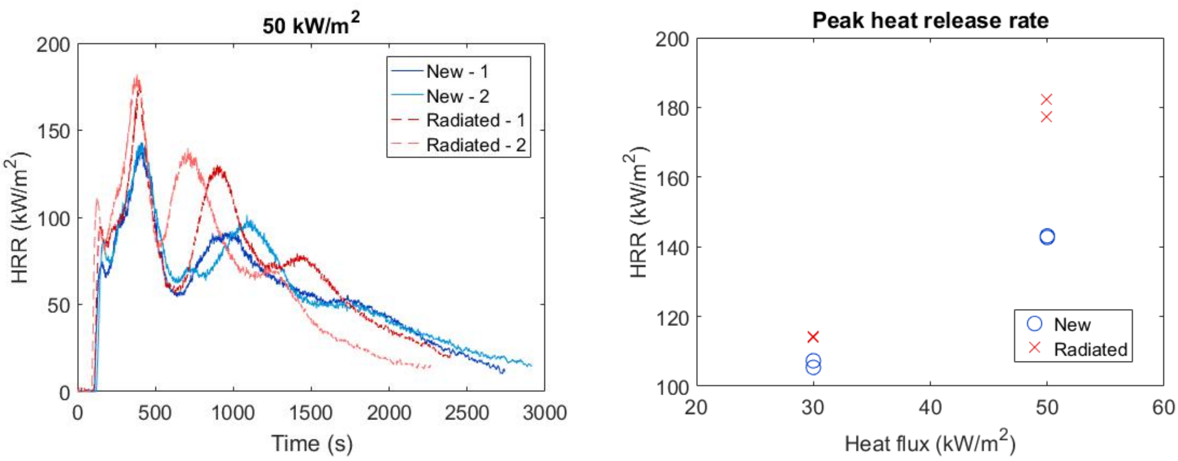
The active tasks during 2016 were:

- WP 1: Cable fire risks during plant life cycle
  - Task 1: New flame retardant polymers,
  - Task 2: Impact of cable ageing.
- WP 2: Fire-Barrier performance assessment
  - Task 1: Barrier performance assessment with Fire-CFD.
- WP 3: Fire simulation development, maintenance and validation.
  - Task 1: FDS development, maintenance and validation.
- WP 4: Participation to PRISME2.

- o Task 1: PRISME utilisation,
- o Task 2: Participation fee.

The work with the new flame retardant polymer continued using reactive molecular dynamics. The simulations with high density polyethylene (HDPE) and a flame retardant Aluminium trihydrate (ATH) were promising. The correct flame retardancy mechanism could be repeated in the simulations, and in addition, an accurate enthalpy of decomposition for ATH could be extracted from the simulations.

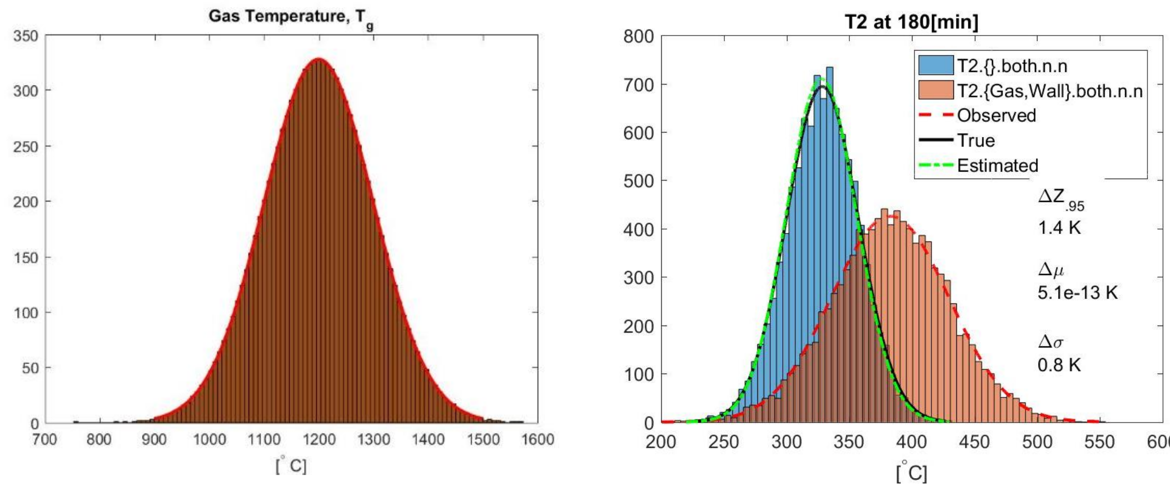
In the research with ageing of cable materials the work continued by studying the combined of accelerated radiative ageing in elevated temperatures. Flame retardant cables were first exposed to relatively high radiation dose during certain time (corresponding to LOCA or about 45 years of service time) in elevated temperatures (75 °C). The cables were then tested in small scale experiments (thermogravimetric analysis, TGA) and bench scale experiments (cone calorimeter), and the results were compared to the non-aged samples. No differences could be observed in the small scale experiments, suggesting that the changes are not due to evaporation or other change in the material composition. However, the cone calorimeter tests showed some small but significant differences between new and aged materials. The most importantly, the time to ignition was slightly decreased with radiated cables, and the peak heat release rate and the effective heat of combustion were slightly increased. Comparison of some cone calorimeter results of the cables is shown in Figure 2.3.2.2.



(a) (b)

Figure 2.3.2.2. Cone calorimeter results of new and radiated cable samples. (a) Heat release rate at 50 kW/m2. (b) Peak heat release rate.

In the barrier performance assessment the work concentrated on the uncertainty propagation between two models. Figure 2.3.2.3a shows the distribution of gas temperature and Figureb shows an example of the true, observed and estimated output distributions, in a case where there is systematic bias and the random error in both gas temperature and heat conduction model {Gas, Wall}. The errors are imposed each iteration of MC simulation. FEM model takes into account both radiative and convective heat fluxes (”both”) and the distribution type selected for both, the input gas temperature and the wall material properties is normal distribution (“n.n”). From the Figure 2.3.2.3b one can notice that the estimated output distribution closely matches the true output distribution. This indicates that the true distribution of the output can be estimated using a simple statistical relationship provided that the quantitative measure of total modelling error is available.



(a) (b)

Figure 2.3.2.3. (a) Gausion profile of Gas temperature, a input parameter for stochastic analysis. (b) True, observed and estimated distribution of cold side temperature, T2 at = 180 [min].

In the FDS development, maintenance and validation task predictive simulations of liquid pool fires in mechanically ventilated compartments were conducted. The results showed that code is capable of predicting the steady state burning rates of the TPH pool fires in in compartments within 15 % of the experimental value. The effect of lowered oxygen vitiation on the burning rate of pool fires was correctly captured. Simulations were performed using the Fire Dynamics Simulator and the experiments considered were conducted in the OECD PRISME project. The main difference between the conducted study and previous simulation studies is the use of a detailed liquid evaporation model and the direct calculation of the vitiation and thermal environment interactions through the CFD solver.

The participation to PRISME 2 was continued. The project provides high quality, large scale experimental data on the topics that are relevant to fire safety of nuclear power plants. These results can be utilised directly in the safety assessments, or for simulating and validating the simulation tools.

Deliverables in 2016

- The results of the MD simulations were reported and discussed in the updated version of the year's 2015 report. Antti Paajanen, Jukka Vaari: Atomistic modelling of novel fire retardants. VTT-R-04781-16.

- The combined ageing effect of radiation and elevated temperatures on electrical cables was studied experimentally via small and bench scale methods. Anna Matala: Ageing of flame retardant cables. VTT-R-05490-16.

- The uncertainty propagation from one model to another was studied numerically using Monte Carlo simulations. A method for calculating the acceptance criterion for the model uncertainty was also developed. Deepak Paudel and Simo Hostikka: Model uncertainty propagation in fire-barrier performance analyses.

- A journal manuscript was submitted and later accepted on the topic of modelling liquid pool fires. Topi Sikanen, Simo Hostikka: Predicting the Heat Release Rates of Liquid Pool Fires in Mechanically Ventilated Compartments. The same paper will be presented in June 2017 in the international conference of IAFSS (the International Association of Fire Safety Science).

- The latest PRISME2 results have been presented to the RG in several RG meetings, and a special ad hoc meeting was organized during spring 2016 to discuss the results and the possibility of joining PRISME3 project during years 2017-2021.

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).

The focus areas are: 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 relaxation in BWR NPPs.

Specific goals, results and deliverables in 2016

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 computational assessment of flaw behaviour due to fatigue or stress corrosion cracking (SCC) under operational loading including residual stresses is important for determining remaining lifetime of components. There is uncertainty concerning how to computationally assess the possible flaw behaviour, particularly in the case of residual stress loads. The numerical computation for assessing the crack criticality and possible growth is not straightforward due to the limitations of traditional methods to evaluate the parameters describing the crack loading. In this work package, a J-integral evaluation procedure suitable for assessing the crack driving force in residual stress fields was implemented in the FE-code Abaqus.

The deliverables of this work package are:

- A J-integral evaluation routine programmed in Abaqus.

WP2 provides an investigation on the susceptibility of BWR RPV internals and their supporting structures to various relevant degradation mechanisms. The four-year 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 (performed in 2016-2017). The computational part covers both deterministic and probabilistic approaches. The main purpose of the WP during the four years is to prepare a dissertation. The 2016 results concern the computational assessment of the propagation of degradation in the susceptible BWR RPV internal and new computational developments for assessment of the propagation of degradation. A draft of the upcoming dissertation is the only deliverable of this WP in 2016.

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 2.3.3.1). 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.

The deliverables of this work package are:

- The results of and a report on strain-controlled fatigue tests in PWR water environment.

- A conference publication on direct strain-controlled fatigue testing in simulated PWR water

- A work plan formulating the next steps in developing the F_{en}-model

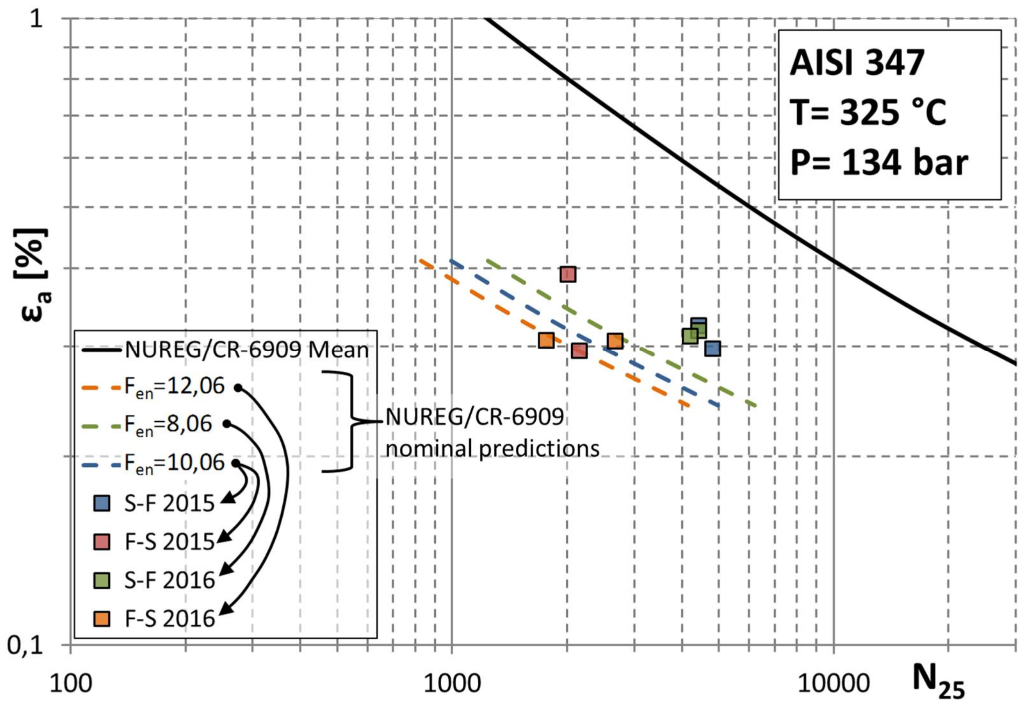


Figure 2.3.3.1. Fatigue test results in PWR water compared to the NUREG/CR-6909 mean curve and nominal expected Fen.

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. Computational fluid dynamics and structural calculations of thermal mixing in a T-junction were performed with the commercial Star-CCM+ and Abaqus codes. In large-eddy simulations (LES), the fluctuating wall heat fluxes obtained with different wall treatments and meshes were compared. In the structural calculations, turbulent thermal loads from LES were applied in 3D modelling while turbulent, spectrum and sinusoidal loads were applied in 1D modelling. Crack growth results using the 1D and 3D models and the different thermal loads were compared.

In addition, the crack growth and component lifetime in Trueflaw's crack manufacturing experiments that use tailored thermal load cycles were studied. The studied case corresponds to low-cycle conditions with considerable cyclic plasticity induced by the thermal load. The results indicated that including the crack in the analysis is needed to obtain realistic crack driving forces as the stresses calculated without the crack resulted in an overestimated growth rates. The use of a crack opening angle crack driving force parameter for low-cycle thermal fatigue was proposed.

The deliverables of this work package are:

- A research report presenting the use of a spectrum method for modelling crack growth due to thermal mixing
- A research report presenting the evaluation of the driving forces and growth of cracks induced by low-cycle thermal fatigue 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 Monte-Carlo or Markov based probabilistic models to evaluate the failure probability and effect of inspections. An example failure probability plot obtained with the VTT RI-ISI procedure is shown in Figure 2.3.3.2. In 2016, the VTT RI-ISI procedure was extended by including corrosion as a degradation mechanism.

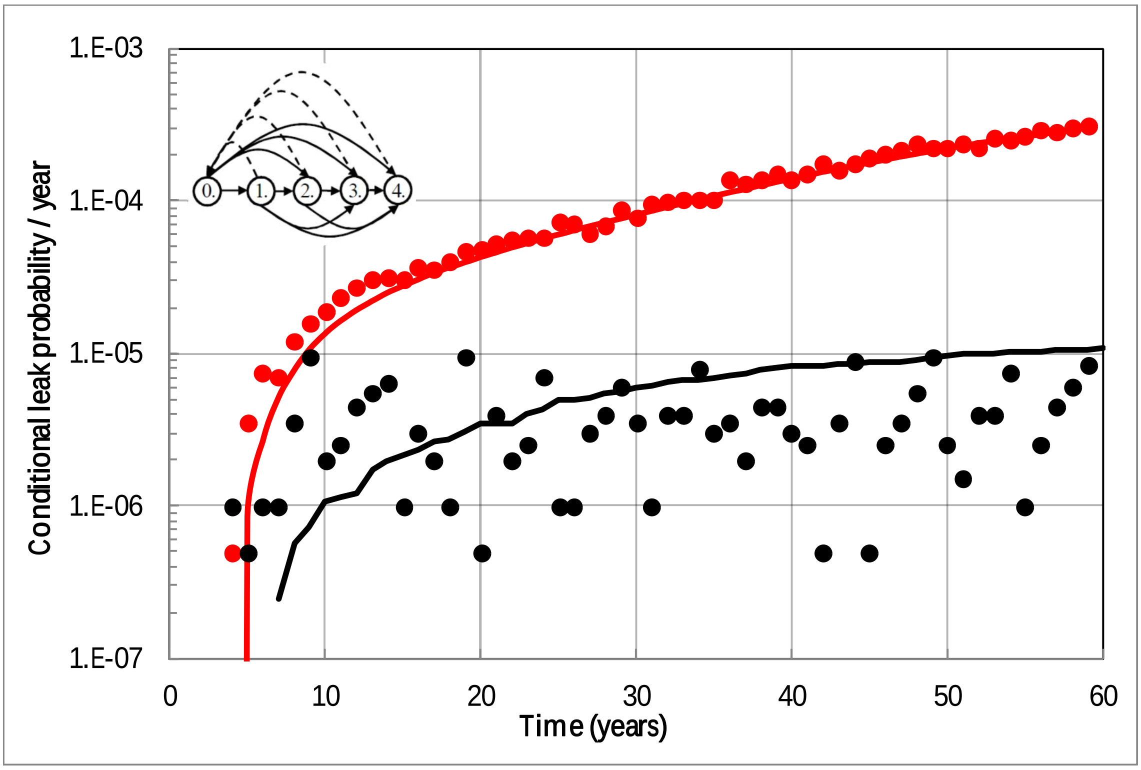


Figure 2.3.3.2. Conditional yearly leak probabilities simulated with the VTT RI-ISI Markov based procedure (lines) and Monte-Carlo based procedure (dots). The red line and dots denote cases without in-service inspections (flow of the ageing process along the solid lines in the subfigure) and the black line and dots denote cases with inspections (flow also along the dashed lines in the subfigure).

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 a new RI-ISI feature, which calculates CCDPs of piping component failures, in PRA software FinPSA. Currently, the RI-ISI feature calculates only CCDPs, but RI-ISI analyses require also failure probabilities of piping components, which may be obtained using the VTT RI-ISI procedures.

The deliverables of this work package are:

- A research report detailing the implementation of corrosion as a degradation mechanism in VTT RI-ISI
- A research report on the computation of consequences of piping component failures in PRA software

WP6 assesses both the dynamic behaviour of piping systems and the loads experienced by restraining pipe supports. The work package develops methods for reducing the computational effort of nonlinear piping analyses. The research includes a review of the most suitable qualification and benchmarking cases for piping analyses. The performed survey is concerned with the qualification processes and benchmark problems for dynamical spectrum and time-history analyses of large-scale piping systems that are subject to typical NPP conditions. Safety-related applications of the design and maintenance standards for an assessment of the structural integrity of piping systems are addressed. Additionally, the piping component benchmarks involving the temperature loads, large deformations and plasticity are included. A summary of selected benchmark problems to exemplify software qualification for mechanical analyses of the large-scale piping systems and separate piping components is included.

The primary stress limit equation in Section III of the ASME B&PV code requires a calculation of a piping resultant moment by combining the moments from simultaneous design loads. The combination procedure for the results from several dynamic time-history analyses may become computationally laborious, depending on the level of required accuracy. This aspect was studied in the project. To reduce unnecessary conservatism, the moment components applied in the summation equation can be selected physically from the same time instants, requiring a search of the maximizing instants from the time step combinations between the dynamic load cases. Combining the moments from time-history analyses was studied by using moments from dynamic piping analyses or by generating random moment signals. It was found that significant computational savings can be achieved by a simple screening procedure to remove the unnecessary time points from the load combination analysis.

The deliverables of this work package are:

- A study of moment combination methods for dynamic analyses
- A review of qualification of nuclear plant piping analyses.

WP7 studies the relaxation of residual stresses. They play a major role in stress corrosion cracking (SCC), which is identified as significant degradation mechanism for various BWR components. Experience from ageing NPPs indicates slower stress corrosion crack growth in many components than would be expected under currently postulated stresses. Residual stress relaxation decreases the effective loads during the service life and therefore slows down SCC. The effect of residual stress relaxation on SCC is not, in general, considered in crack growth calculations although thermal and mechanical loads are known to relax residual stresses significantly. This is due to insufficient data available on the stress relaxation.

In the project, residual stress relaxation in BWR NPP was studied with co-operation of Aalto University and Teollisuuden Voima Oy. Previously used measurement methods are developed further, most notably the contour method. The spatial resolution of the contour method was significantly improved by adopting white light interferometry measurement to the cut surface. This development also vastly increased the amount of measurement data to be analysed, and necessitated significant development of the measurement pipeline.

The residual stresses measured from the pipe sections removed from OL1 and OL2 are compared with the other experimental data. A removed T-junction was provided for the residual stress measurements. Prior to measurement, the component had been cut in two parts. This was done for other investigations, but also provided measurement access to inner surface of the T-junction. The components had also been decontaminated prior to measurement. The sections were measured using X-ray diffraction and the contour method (using white light interferometry and the newly developed measurement pipeline). Contour measurements were completed on four sample surfaces. Typical results are shown in in Figure 2.3.3.3.

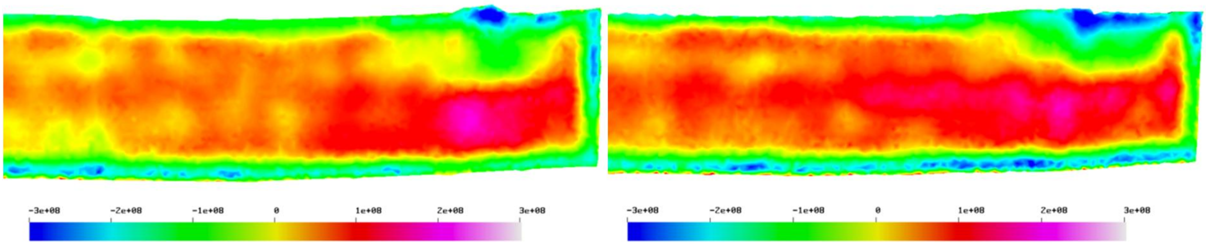


Figure 2.3.3.3. Circumferential residual stress contour measurements from two weld samples (stress in Pa).

The results indicate that the service loads have caused some relaxation of residual stresses in the weld. The stresses are somewhat lower than were measured for the reference pipe sections. Also, the tensile stress region is confined more clearly inside the weld and there is a balancing compressive stress in the weld root area.

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

2.3.4 LOST - Long term operation aspects of structural integrity

The general objective of the long term operation aspects of structural integrity (LOST) project is to develop methods and tools for structural safety analysis of primary circuit components, reactor pressure vessel (RPV) and dissimilar metal welds (DMW). In 2016 the project covered tasks related to fast fracture in the upper shelf area for RPV like materials, experimental and numerical work on dissimilar metal welds (DMWs) and thirdly, international cooperation. In 2016, related to fast fracture, the first experiments of the extensive experimental program were done. Related to DMWs numerical and experimental methods were used to investigate crack growth. In addition, residual stress fields in DMWs, after repair, were also investigated. Two scientific articles were written and two conference papers, in addition to several reports. Also investigations on the Barsebäck 2 reactor pressure vessel weld were done.

Goals and results in 2016

One objective in 2016 was to make J-R tests at different temperatures to investigate the effect of fast fracture events in the upper shelf area. The research topic is justified by the requirements in the Finnish regulatory requirements that state the following; In connection with the strength analysis of Safety Class 1 pressure equipment, an assessment shall be given on the potential for a fast fracture occurring in the upper shelf area where temperatures exceed the transition temperature zone. Physically, upper shelf is defined as the temperature range where brittle fracture cannot occur. Probability of fast fracture in upper shelf area during operation conditions is small, due to the high tearing resistance of the RPV material. It is more probable that fast fracture could occur in thick-walled components which undergo rapid cooling under high pressure. Therefore, investigations of tearing resistance under transient temperatures are important.

The results from 2016 show that the tearing resistance decreases from 25 to 200 °C, but increases from 200 to 300 °C. Additionally, one test series has been done in the room temperature regime with the multi-specimen test method, to get a better understanding of the differences to single specimen test techniques and increasing the accuracy of the single specimen testing procedure. Related to this work package, advanced structural integrity, the objective for the whole SAFIR2018 period is to develop new advanced structural integrity methods to describe the ductile crack growth during a temperature transient accounting for temperature history effects. The upcoming work consists of experimental investigations of ductile tearing resistance during transient temperatures.

Related to advanced structural integrity assessment methods, one objective in LOST is to use miniature size C(T) specimens in determination of ductile-to-brittle transition temperature. This new method reduces the material required for surveillance testing. In 2016 miniature sized specimens were validated for a homogeneous material. It was concluded that the initiation location in miniature sized specimens has a similar distribution as standard sized specimens. In addition, the miniature sized specimens were validated for a RPV weld material. The tests on the weld material with miniature sized specimens yield similar results as standard sized specimens. Previous investigations have been mainly done on homogeneous materials. Yet, further investigations are required for the use of miniature sized specimens for welds. This is done by investigating the applicability of miniature sized specimens for irradiated welds. The investigated weld was from Barsebäck 2 RPV and the research activities are related to BREDA project, a Scandinavian collaboration project.

Another goal was to investigate the local stress-strain properties of near interface zones (NIZs) and the effect of the strength of these zones on tearing resistance in welds and DMWs. The interface region of dissimilar metal welds (DMW) consists of narrow microstructural zones with varying fracture mechanical and mechanical properties. The narrowest zones can be just few micro meters wide and between adjacent zones there can exist a major difference e.g. in strength properties. As the material fractures the properties of these adjacent and other zones interact. This interaction affects the tearing resistance. Even if the fracture occurs in a zone with nominally high tearing resistance, the mechanical properties of the adjacent zones can lower the tearing resistance.

In 2016 eta factors for the interface region of DMWs were derived in LOST, with accurate information of mechanical properties of the narrow zones in the interface region. After the eta factors have been derived with accurate models, there is knowledge of regions where homogeneous solutions can be applied and of regions where more precise eta factors are needed. However, the numerical model used in 2016 did not produce reliable eta factors for DMWs and in 2017 another type of numerical model is used to derive these factors.

Another issue related to this heterogeneity in the interface region is measurements of tearing resistance of dissimilar metal welds. The plastic strain in front of the crack tip can be constrained to a small region. In a homogeneous specimen of the same material, the strain would be constrained in a larger region. The constraint situation is also dependent on specimen configuration. Results from 2016 show that constraint has an effect on tearing resistance. The closer the constraint of the test specimen is to the constraint of components the larger is the tearing resistance. In standard specimens, the constraint is larger and therefore, the tearing resistance is smaller. This means that the use of standard sized specimens to determine tearing resistance produce conservative results.

Yet another gaol was to publish two scientific articles. One article was based on experimental work on DMWs and in the other work a numerical model was used to simulate crack growth in DMWs. The first paper was based on results from European research projects MULTIMETAL and relies also on knowledge gained from BIMET, ADIMEW, STYLE, PERDI and National research projects SINI and FAR. The experimental article discussed tearing resistance curves measured with 10x10 and 10x20 SE(B) specimens and the effect of initiation location on tearing resistance. The initial crack was located in the interface region between a ferritic steel and weld metal Inconel 52 of a DMW. The main conclusion is that for DMW HAZ cracks that deviate to the fusion boundary tearing resistance is higher for the cracks initiating further away from the fusion boundary. This result affects how the lower boundary values of DMWs shall be identified. Because the initiation location in the heterogeneous interface of DMWs has a big impact on tearing resistance, a standard developed for heterogeneous materials shall contain guidelines on microstructural characterisation of the fracture initiation location.

The other article was based on numerical simulations on crack growth in heat-affected zone of a VVER dissimilar metal weld. The main conclusion is that to model crack growth in HAZ of DMWs the thickness of the different material zones in HAZ has to be optimized. The optimization is required to capture the effect of the gradual strength changes in HAZ on crack growth behaviour in the numerical model. The strength changes in HAZ occur over few mm.

The objective for the whole period of the SAFIR2018 programme (related to DMWs) is to develop advanced tools for material characterisation of DMWs in terms of fracture toughness testing with associated J-integral calculation and the quantification of crack deviation driven by local strength mismatch state. The results achieved in 2016 support these objectives. Firstly, an advanced numerical model was used to predict the crack growth path in DMWs. Secondly, the effect of crack location on fracture toughness in the interface region between two materials was quantified. Thirdly, the effect of specimen configuration on fracture toughness was investigated, and revealed that J-R curves obtained with current standardised fracture toughness methods are conservative.

A final objective related to DMWs was to investigate the effect of residual stresses. Residual stresses in DMWs that have been thick overlay and inlay welded were investigated. The results show that the decrease of the circumferential stress due to overlay welding is small at depths larger than half of the wall thickness from inner surface. The situation is the same in the case of combined inlay and overlay welding. On the basis of literature, weld inlay has been successfully applied in several cases. The good stress corrosion resistance overrides the un-favourable effects of the high tensile stresses caused by the inlay welding.

The purpose in the international cooperation task was to take part in ASTM Committee E08 Fatigue and Fracture executive committee meeting in May and November. In 2016 Research Professor Kim Wallin participated in these two meetings. In the meetings, the current fracture mechanical methods are improved, which increases the accuracy of the methods. This has an impact on nuclear safety. Fracture mechanics is used in NPPs to define the tolerable crack size exploited in estimation of safe continued operation of the NPP and in making decision about component replacement.

Deliverables in 2016

- A research report on the stresses due thick overlay welding. The overlay welding reduces circumferential stresses especially near the inside surface. The results show that the decrease of the circumferential stress due to overlay welding is small at depths larger than half of the wall thickness from inner surface. The situation is the same in the case of combined inlay and overlay welding.

- A research report on inlay welding as a nozzle repair method, literature survey of the residual stress computations. The main outcome; on the basis of literature weld inlay has been successfully applied in several cases. The good stress corrosion resistance overrides the un-favourable effects of the high tensile stresses caused by the inlay welding.

- A research report on “BREDA: Fracture toughness measurements with miniature C(T) specimens in reference condition”. The results show that T0 for the Barsebäck RPV weld in reference condition is around - 100 °C, which is in alignment with previous results. This work is a pre-investigation for the Barsebäck trepan testing project carried out in 2018-2019.

- A research report on tearing resistance analysis of a VVER dissimilar metal weld mock-up. The main result is that the conventional fracture toughness equations can produce non-conservative results compared to fracture toughness estimates with DMW specific equations.

- A research report on dissimilar metal welds – the effect of crack path and specimen configuration on tearing resistance. Tearing resistance is higher for HAZ cracks that initiate further away from the fusion boundary, even in cases where the crack deviates to the fusion boundary. Furthermore crack path analyses of SE(B) specimens (NG DMW Inconel 52/18MND5) show that if the initial crack is 20 um on weld metal side, then the tearing resistance is higher than at the fusion boundary. Based on these results more precise and realistic methods can be developed to ensure the safety of NPP pipes/DMWs.

- A research report on crack growth computation in dissimilar metal weld joints by local approach. J-integral curves for the growing cracks were computed based on ASTM E1820-13, good fits were found between numerical and experimental J-integral curves.

- Contribution to ASTM Committee E08 Fatigue and Fracture executive committee in May 2016 and November 2016. Kim Wallin participated in two ASTM meetings in 2016.

- Conference article accepted in the ECF21 conference, based on fracture mechanical characterisation of dissimilar metal welds (DMWs). Outcome: characterisation of crack propagation location in the interface region is crucial and shall be included to guidelines on experimental characterisation of DMWs.

- Conference article accepted in the conference of the ASME 2016 Pressure Vessels and Piping Conference PVP2016. The article investigates the location of initiation sites in fracture toughness testing specimens and the effects of size and side grooves. The results show that the initiation location as a function of the centre of specimens is the same in conventional large specimens as in miniature sized specimens.

- A journal article on numerical simulations on crack growth in heat-affected zone of a VVER dissimilar metal weld. The main conclusion is that to model crack growth in HAZ of DMWs the thickness of the different material zones in HAZ has to be optimized. The optimization is required to capture the effect of the gradual strength changes in HAZ on crack growth behaviour in the numerical model. The strength changes in HAZ occur over few mm.

- A journal article on the effect of crack path on tearing resistance of a narrow-gap Alloy 52 dissimilar metal weld. The main conclusion is that for DMW HAZ cracks that deviate to the fusion boundary tearing resistance is higher for the cracks initiating further away from the fusion boundary. This result affects how the lower boundary values of DMWs shall be identified.

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 2016

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. Identifying a replacement of hydrazine is an important target for this work. In 2016, a literature study was performed on this issue.

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 feed water line corrosion is a major goal in this study. In 2016, a series of experiments was conducted to determine the effect of a particular film forming amine, octadecylamine (ODA) on the rate of corrosion, i.e. the rate of formation of magnetite particulates.

Another clearly established cause of SG corrosion damage is the lead assisted stress corrosion cracking, PbSCC. In 2016 an experimental arrangement for studying PbSCC was developed and verified by a short series of experiments.

International co-operation was continued in 2016 e.g. in the form of a visiting scientist from Manchester University in UK, Mr. Max Szolcek, who was working at VTT between 25.3 – 14.5.2016, participating mainly in experiments on determination of surface charge of magnetite.

Mechanism of lead assisted stress corrosion cracking of carbon steel

The magnitude of corrosion damage around the SG tubes is typically enhanced by enrichment of impurities within crevices. This enrichment is driven by boiling, where water entering the crevices within a steam generator (e.g. between tube and tube-support or under a magnetite deposit on a straight tube) boils, allowing volatile species to escape as steam and leaving behind non-volatile species (salts, lead, copper etc.) in the small water volume of the crevice. 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 shown to be susceptible to PbSCC.

The technique for studying SCC in this project is 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 or the same environment but without the SCC promoting agent) and additionally from the morphology of the fracture surface.

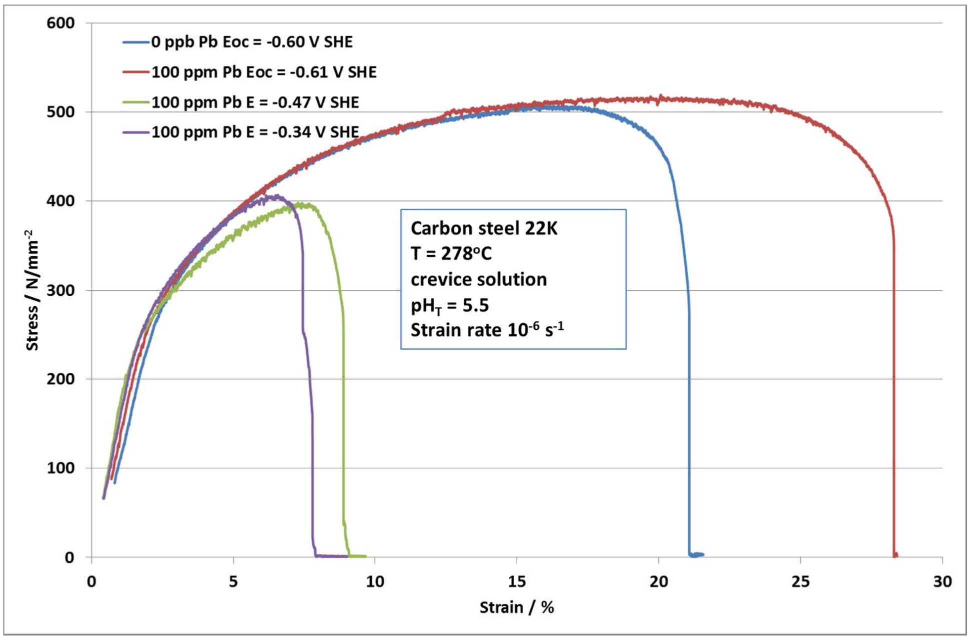


Figure 2.3.5.1. Comparison of stress-strain –curves of carbon steel 22K in crevice solution at T = 278^{o}C with and without Pb, at corrosion potential and at slightly elevated potentials.

Table 2.3.5.1. SEM/EDS analysis results, surface of the SSRT-specimens, weighted average of oxygen, iron, sulphur and lead, a-%.

Test	*O*	*Fe*	*S*	*Pb*
1	52.3	47.4	0.3	0
2	55.8	40.2	1.2	2.8
3	49.5	50.4	0.1	0
4	62.6	37.2	0.1	0.1

Figure 2.3.5.1 shows a comparison of stress-strain curves of of carbon steel 22K in crevice solution at T = 278°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 below 10%.

Electrochemical impedance spectroscopy (EIS) measurements revealed that in presence of 100 ppm Pb at corrosion potential there is practically no passive film on the surface of carbon steel 22K, i.e. the surface is in an active state and therefore any localised corrosion mode such as e.g. SCC becomes impossible. Ex situ studies of the exposed surfaces confirmed the presence of Pb on the surface at corrosion potential, but not at elevated potentials, Table 2.3.5.1. Current-voltage curves (i.e. polarisation curves) confirmed that the effect of Pb is to activate the surface and that at slightly elevated potentials a semi-passive surface film is formed on it. Thermodynamic phase stability calculations (Pourbaix-diagrams) revealed that at potentials about 0.1 V higher than the corrosion potential, Pb can dissolve from the surface as $PbCl^+$, in accordance with the SEM/EDS results shown in Table 2.3.5.1.

The main conclusion from this work is that a combination of a mechanical testing method, in this case the SSRT test method, in situ electrochemical methods and ex situ surface analytical methods are needed to gain further insight into the mechanism of PbSCC.

The system validation tests with carbon steel 22K in a representative slightly acidic crevice environment were rather successful. Based on the results, the effect of Pb is to activate the surface so that any local corrosion mode, such as SCC, becomes practically impossible, and general corrosion is observed instead. However, if the environment changes to more oxidising one by as little as 0.1 V, Pb dissolves from the surface possibly as $PbCl^+$, leaving the surface in a semi-passive state and very susceptible to SCC.

Effect of octadecylamine (ODA) on PWR feed water line corrosion

There are three main routes to mitigate the corrosion problems caused by magnetite deposition on SG surfaces. The first one is to modify the water chemistry so that the source term of magnetite particles, i.e. corrosion of carbon steel components along the feed water line is minimised. This can be achieved e.g. by injecting a film forming amine into the secondary side loop or by controlling the secondary side water pH to be between 9.6 and 10, which coincides with the minimum in iron dissolution rate and thus minimises the carbon steel corrosion rate. The second route is to select the water chemistry so that the magnetite particles keep in colloidal form and can be removed by filters before they have time to deposit into the SG. This can be done by adding a dispersant (such as polyacrylic acid, PAA) or by selecting a suitable combination of amines for the pH control. The third route is to prevent the detrimental action of the already existing magnetite deposits. This can be done by removing the deposits during outages frequently enough or by introducing crevice inhibitors (such as TiO_2 or a film forming amine).

Film forming amines (FFA) have been found to be efficient in mitigating several of the detrimental aspects related to magnetite deposits. FFAs effectively reduce the source term, i.e. feed water line component corrosion by more than 90%, even at elevated pH close to 9.8. In addition, FFAs have been shown to be able to mitigate crevice corrosion, i.e. decrease the aggressiveness of existing crevices within SGs. As FFAs have so far been tried only in two PWR plants and one CANDU plant, (and of course in VVERs and conventional plants), there is a need for further studies on their application.

In the current project the effect of octadecylamine (ODA), one of the FFAs, on the corrosion rate of carbon steel and copper is studied using exposure coupons and in situ EIS as the main tools. Figure 2.3.5.2 shows a comparison of impedance spectra measured for carbon steel 22K at T = 228^{o}C in ammonia only (three repetitions) and ammonia with 20 ppm ODA (four repetitions). The impedance magnitude is about x3 higher with ODA, indicating roughly three times lower corrosion rate. In case of copper a similar x3 reduction in corrosion rate was observed. Reduction in the corrosion rate by roughly a factor x2 was confirmed also using conventional corrosion coupons exposed to the environment.

ODA is a film forming amine that can be used in the secondary side water of PWRs to mitigate corrosion within the feedwater line and thus also reduce magnetite deposition into steam generator, mitigating corrosion probelems in the SG. ODA addition was found to have similar effect on the carbon steel corrosion rate whether it was added at the beginning of the exposure (fresh surface) or after a period of pre-oxidation of the carbon steel surface during which a stable magnetite film was grown. This indicates that ODA would possibly slow down also the corrosion of carbon steel at locations where flow assisted corrosion (FAC) takes place.

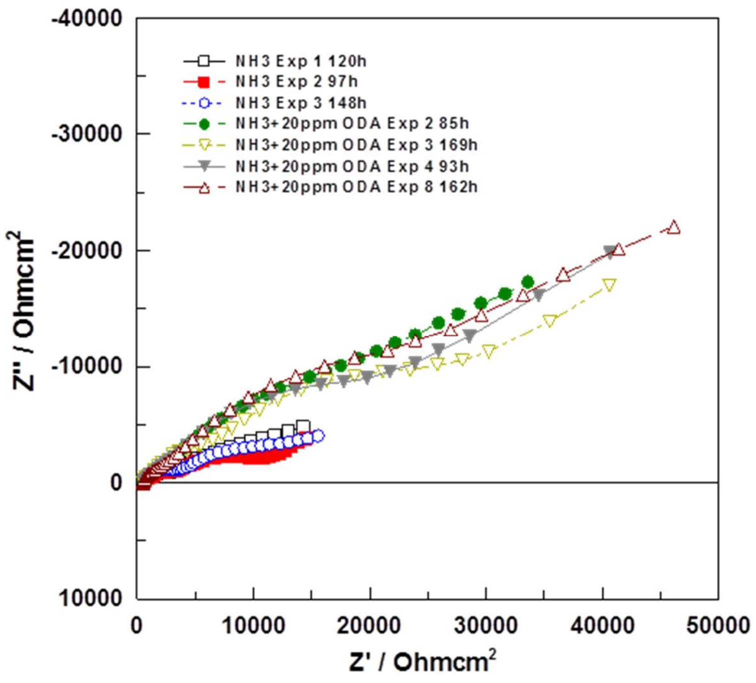


Figure 2.3.5.2. Impedance spectra of carbon steel (22K) at T = 228^{o}C in secondary side water treated with ammonia to pHRT = 9.2 to 9.8. Comparison of three repetitions without and four repetitions with 20 ppm (3.3 ppm in water phase) added ODA as emulsion.

The main disadvantage of application of a film forming amine is that the on-line monitoring tools, e.g. conductivity sensors, unless withdrawn for the FFA application period, will also be covered with the thin FFA film and possibly lose their reliability. Removing and re-assembling the sensors before and after each FFA application campaign represents a notable effort which has to be considered when estimating the benefits of an FFA.

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

The objective of the joint VTT – Aalto THELMA project is to support the safe operation of NPP´s through increased understanding of the influence of light water reactor environments on the behaviour of nuclear materials, with the emphasis on thermal ageing. To meet these goals, several tasks are pursued dealing with thermal ageing of stainless steel cast material and weld metals and of wrought Alloy 690 material. The crack initiation susceptibility of nuclear materials is investigated, focussing on developing an accelerated method for testing, while the latest research results on corrosion fatigue assessment are brought to the programme through participation in an EU-project and reporting the results within THELMA. Our capabilities to perform initiation testing in simulated PWR environment is bench-marked through participation in a Round Robin exercise. International co-operation is important as a tool to bring the latest knowledge to Finland and benchmark the scientific level of our own research. Knowledge transfer and continuous education will secure uninterrupted availability of high-quality expertise for ageing management.

Specific goals in 2016

The THELMA 2016 project has several goals dealing with the long-term behaviour of nuclear materials.

The long-term behaviour and ageing mechanisms of 316L type welds are investigated using thermally aged materials up to 40 000 h in co-operation with MIT, USA. The overall results show hardening of the δ-ferrite, measured by nano-indentation hardenss, and formation of G-phase at higher ageing temperatures (400 °C), degradation of mechanical properties (impact energy and fracture toughness) and especially of the fracture toughness in environment. Further, the crack growth rate in BWR conditions is about double for a severely aged material compared to a non-aged material. The crack path in the aged material follows preferentially the ferrite-austenite phase boundary and show indications of a hydrogen related mechanism.

The thermal ageing of cast stainless steel of type CF8M is investigated in co-operation with doctoral student M. Bjurman from KTH, Sweden. The materials, delivered to the project by the supervisor, professor Pål Efsing, Vattenfall and KTH, are from the cold and hot leg from a steam generator and have been thermally aged for 70 000 h at 291C and 325C, respectively. In 2016, nano-indentation hardness and electrochemical measurements were performed and results were presented at an international conference. The nano-indentation hardness increase with ageing, and the level is similar to that in thermally aged 316L weld metal, Figure 2.3.6.1. Weak mottled appearance was observed in the δ-ferrite after DL-EPR measurement.

The characteristics and properties as well as the thermal ageing mechanisms of Alloy 690 materials (totally 36 conditions) are determined using thermally aged materials received from KAERI, Korea. Thermal ageing of Alloy 690 triggers an intergranular (IG) carbide precipitation and is known to promote an ordering reaction causing lattice contraction. Variations in hardness and lattice parameter were attributed to the formation of short-range ordering (SRO) in all four conditions investigated with a peak level at 420 °C, consistent with the literature, Figure 2.3.6.2. Prior heat treatment induced ordering before thermal ageing. At higher temperatures, stress relaxation, recrystallization and α-Cr precipitation were observed in the cold-worked samples, while a disordering reaction was inferred in all samples based on a decrease in hardness. IG precipitation of $M_{23}C_6$ carbides increased with increasing ageing temperature in all conditions, as well as diffusion-induced grain boundary migration.

Irradiated stainless steels are characterised for the Halden programme. In 2016, a 2 dpa material from a creep test, showing contraction of the material in the beginning of the test was characterised using transmission electron microscopy. The results showed loops typical for irradiated SS, and some precipitates originating from materials manufacturing. However, no features explaining the observed contraction were found.

The latest new test results and development of assessment methods for environmentally assisted fatigue are made available to the SAFIR2018 programme through participation in the EU-INCEFA+ project. In 2016, most of the materials have been characterised from test specimens, the grain sizes and roughness have been measured. This information will be utilised in the test data evaluation.

An accelerated crack initiation test method is developed in co-operation with European laboratories taking part in the Nugenia MICRIN+ project. In 2016, testing using tapered specimens and nickel-based Alloy 182 was performed. The results on Alloy 182 showed a decreasing apparent initiation stress threshold vs strain rate. The specimens were flat specimens, with one surface ground and the other polished, which resulted in a higher apparent stress threshold in BWR conditions, i.e., showing the detrimental effect of surface grinding on SCC crack initiation. The effect of surface condition was not as evident in simulated PWR conditions.

Data on crack growth rates of Alloy 182 in simulated BWR conditions were reviewed and collected from the open literature and delivered to the SAFIR2018 FOUND project for further analysis.

The VTT initiation testing capabilities are bench-marked by participation in a Round Robin using Alloy 600 as test material. Two tests were performed in 2016, both at a stress level 0f 0.9 times the yield stress. No initiation was observed at this stress level.

Deliverables in 2016

- Scientific article on thermally aged weld metals summarising the work performed on these materials, including mechanical testing, microstructural characterisation and performance in environment.

- Scientific article on thermal ageing of alloy 690 summarising most of the existing results on thermally aged Alloy 690 using versatile methods, i.e., X-ray, SEM/EBSD, nano-indentation hardness and transmission electron microscopy.

- Manuscript ready for a dissertation on thermal ageing of Alloy 690 by Roman Mouginot, Aalto university. The dissertation will take place in Spring, 2017.

- Conference presentation on results from investigations on thermally aged cast SS including TEM-results performed at VTT.

- Deliverable report for the NUGENIA+ MICRIN+ project, summarising initiation test results on Alloy 182 in BWR environment.

- Report and NUGENIA position paper on initiation testing of Alloy 182, summarising the results from two autoclave tests in simulated BWR environment and summarising existing knowledge and future research needs.
- Conference publication summarising the TEM-observation on 2 dpa irradiated 304L stainless steel tested in-core in the Halden reactor.
- Research report on the progress in 2016 in the EU-INCFEA+ project dealing with environmentally assisted fatigue assessment.
- Research report summarising the results from two initiation tests in simulated PWR environment using Alloy 600.
- Data on crack growth rates of Alloy 182 in BWR conditions was reviewed, collected and delivered to SAFIR2018 FOUND project for further analysis.

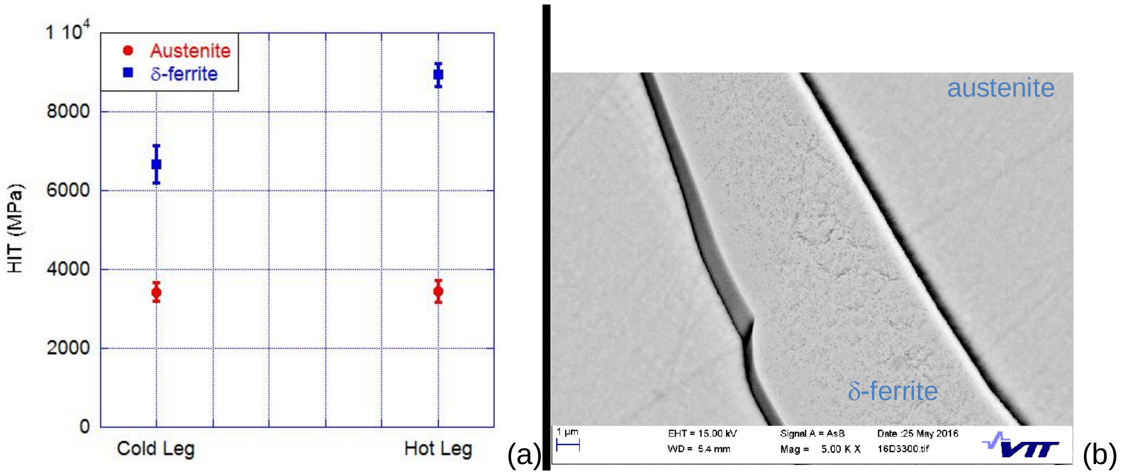


Figure 2.3.6.1: Nano-indentation hardness (HIT) results for the cast stainless steel hot leg, aged 70 000 h at 325 °C, and cold leg, aged 70 000 h at 291°C austenite and δ-ferrite phases (a) and SEM-image of the hot leg material showing weak mottled structure in the δ-ferrite phase.

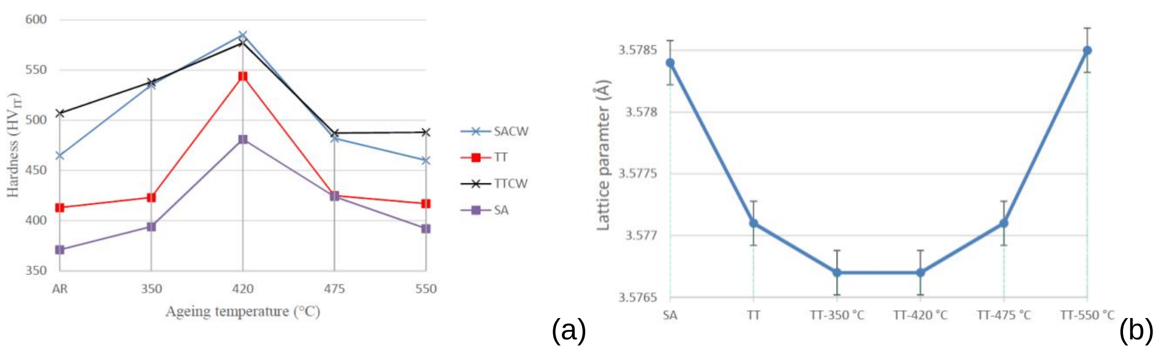


Figure 2.3.6.2: Nano-indentation hardness (a) and lattice parameters, determined using X-ray diffraction, of as-received SA and TT samples, and of the TT condition after ageing at 350, 420, 475 and 550 °C for 10 000 h.

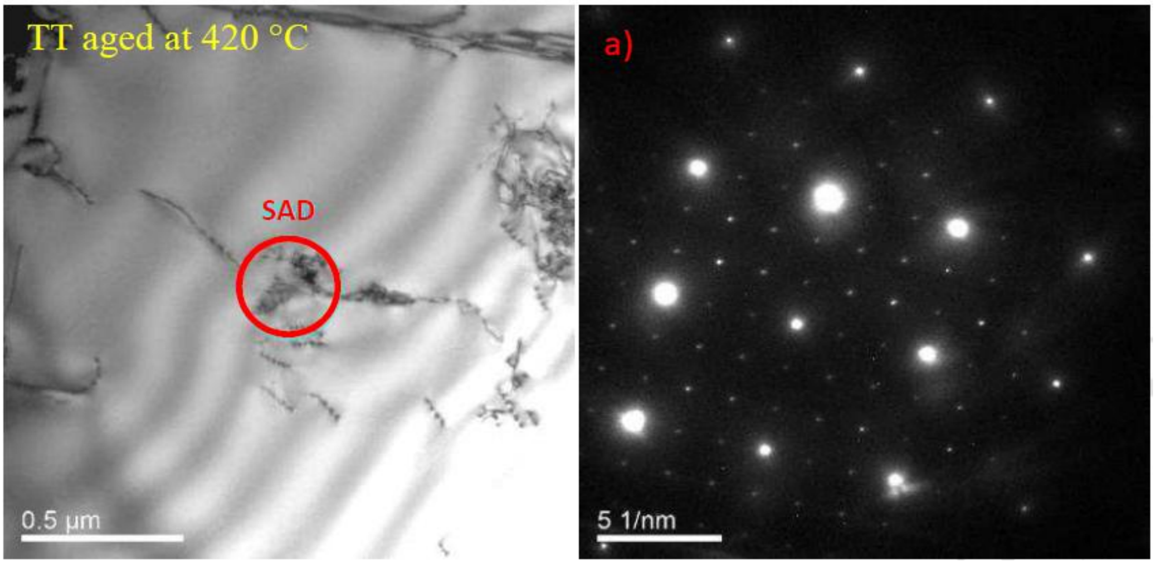


Figure 2.3.6.3: TEM imaging of Alloy 690 thermally treated condition aged for 10 000 hours at 420°C showing the selected area diffraction typical for M23C6 carbides (a) and dark field image associated with the diffraction points (b)

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

A profound understanding of the reliability of non-destructive examination (NDE) methods is needed for safe operation of nuclear power plants (NPP). The project of NDE on NPP primary circuit components and concrete infrastructure (WANDA) applied to the SAFIR2018 programme is focusing on the development and understanding of NDE methods. WANDA consists of two work packages. Work package 1 (WP1) addresses NDE on NPP primary component materials and WP2 focuses on the NDE of NPP concrete infrastructure.

The ISI for primary circuit components is mostly performed in a short time period with limited accessibility. NDE techniques are the main tools to inspect the structural integrity of the primary circuit components in the NPP. The development of the NDE techniques towards more reliable and efficient ISI promotes the safety of NPPs. Artificial defects are typically used as a reference when the performance of an NDT procedure is demonstrated. Because of the lack of real defects, artificial defects are needed for certification and training of the inspectors. According to the previously performed studies in MAKOMON project in SAFIR2014 on artificial defects, the ultrasonic response varies with the type of the defect and with the technique used. To be able to evaluate the severity of the detected defects, it is highly important to know the exact type of the artificial defects in the reference samples and their correspondence to the actual defects. The use of artificial defect can lead to an error, if the limitations of the artificial defect used for the NDE procedure design or qualification compared to ISI -defects (e.g. stress corrosion crack) is not known. Also in recent years, there has been increasing need to better quantify the expected performance and, in particular, to obtain quantitative data on POD for the used inspections. This information is needed, for example, to better facilitate risk-informed in-service inspection.

Spesific goals in 2016

One of the main focuses of the WANDA project is to maintain the expertise level of Finnish NDE research of the NPP component materials and to raise that of NDE of concrete infrastructure. Also one of the important factors for the future is the transfer of know-how in the area of NDE to a younger generation of scientists. Objectives also are: to analyse and further verify the reliability of NDE simulations by evaluating the structure of the austenitic weld and comprising a valid model, to further develop emulation of flaws from low number of artificial flaws in order to evaluate POD, to design and construct a mock-up of a representative reinforced concrete element from a NPP, to pursue international co-operation regarding NDT&E inspections and monitoring systems of NPP RCS.

One Master’s theses regarding the subject of artificial flaws in austenitic stainless steel was finished in the beginning of the year 2016. The master’s theses discussed about simulation of artificial flaws in austenitic stainless steel and also how the structure of the weld effect the detectability and also the simulation results.

Evaluating a POD curve from fewer amount of artificial flaws has been researched. This would lower the cost of deriving a POD curve and give the possibility to evaluate and asses different cabapilities of NDT techniques and different variables affecting the inspection. Also, determing most efficient inspection periods with combination to crack growth simulations might be viable in the future. This yielded a scientific article to be published during 2017.

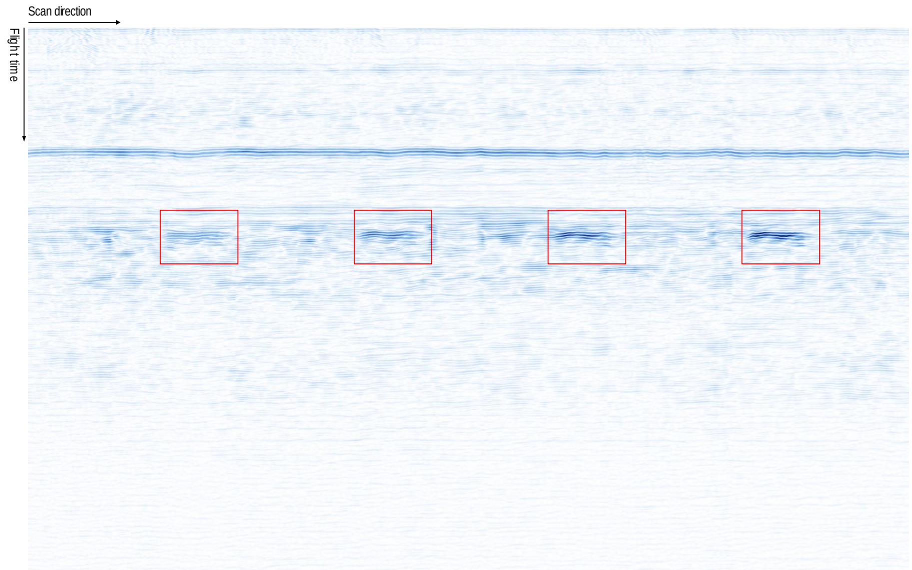


Figure 2.3.7.1. View along the weld with scaled cracks introduced to the data (red rectangles).

The effects of potential deterioration of NPP concrete structures, systems and components (SSC) must be assessed and managed during both the current operating license period as well as subsequent license renewal periods. Reinforced concrete structures (RCS) differ to the many other mechanical and electrical components, as the replacement of these is impractical. Therefore it is clear that the safety issues related to plant aging and continued service of the concrete structures must become thoroughly resolved. The inspection of NPPs concrete structures present challenges different from those of conventional civil engineering structures. As a result, there is a need for NDE of RCS to be able to undertake compliance testing, collection of specific data or parameters, condition assessments, and damage assessment.

The NDE of concrete structures was divided into two tasks. First task was the review of NDE methods and monitoring of concrete performance included the preparation of three literature reviews summarizing the current use of NDE methods, sensors and monitoring systems for RCS, and a summary of NDE research on reinforced concrete with relevance to NPP. The second task was To determine the design requirements for a NPP mock-up concrete specimen, several meetings were held with the Finnish regulator STUK and operating utilities

(TVO and FORTUM), and also with The Finnish Transport Agency who is responsible for the maintenance of Finland's transport system, to roadmap relevant aspects related to NDE of NPP RCS.

There is still a clear need for NDE methodologies to continue to evolve. Research has shown the need for realistic specimens should be developed to allow direct comparisons between various techniques, with consideration given to ensuring a broad range of defects, and to ensure the probability of detection for a method can be properly determined.

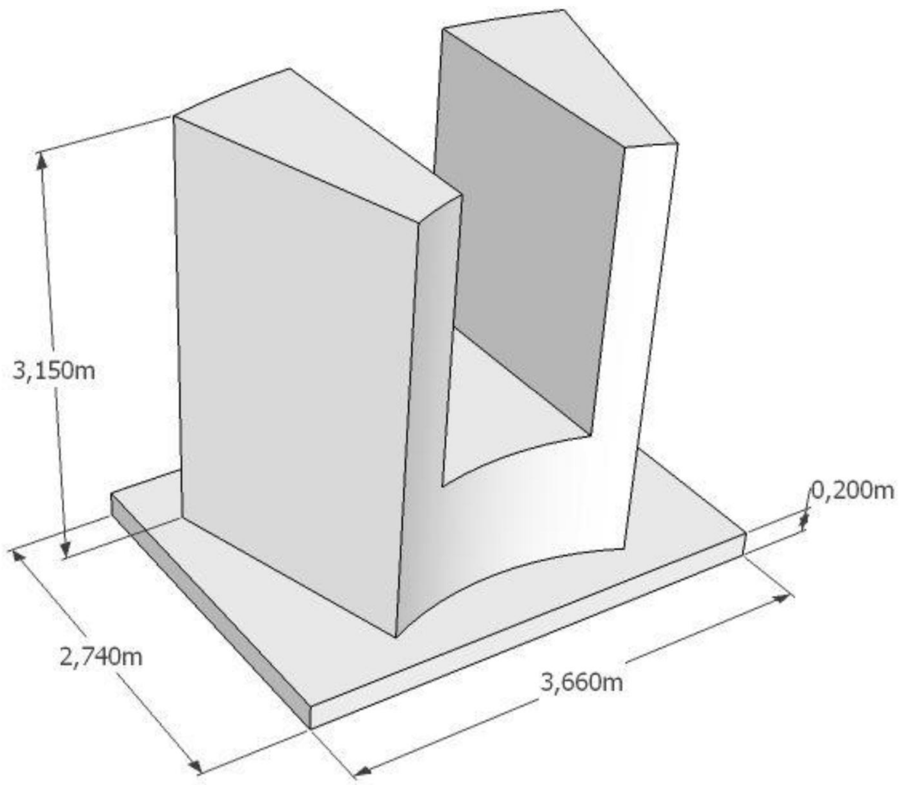


Figure 2.3.7.2. Schematic overview of the TVO -thick-walled concrete structure

Deliverables in 2016

- One research institute report, WANDA VTT-R-00645-17 Pre-design considerations for a large-scale NDE mock-up SAFIR 2018 – WANDA Project WP2 2016 Deliverable, Al-Neshawy, Fahim
- One scientific journal article has been written (Publish date during the year 2017)
- Conference article in Baltica X, Artificial Flaw Detection with Ultrasound in Austenitic Stainless Steels, Koskinen, Tuomas
- Conference article in WCNDT, Ultrasonic Response on Artificially Produced Fatigue Cracks in AISI 321 Austenitic Stainless Steel Weld, Koskinen, Ari
- Conference article in WCNDT, Measurements of the Extension of the Magnetite Pile on Steam Generator Tubing with Eddy Current Techniques, Jäppinen, Tarja

- Conference article in WCNDT, Selection Matrix for Non-Destructive Testing of NPP Concrete Structures, Al-Neshawy, Fahim

- Conference article in WCNDT, NDE Research of Nuclear Power Plant Primary Circuit Components and Concrete Infrastructure in Finland, Jäppinen, Tarja

- Conference article in 12th International Conference on Non Destructive Evaluation in Relation to Structural Integrity for Nuclear and Pressurized Components, The effect of an austenitic weld to probability of detection of ultrasonic inspection, Koskinen, Tuomas Conference article in 12th International Conference on Non Destructive Evaluation in Relation to Structural Integrity for Nuclear and Pressurized Components, PAUT Sizing Artificially Produced Fatigue Cracks in Austenitic Stainless Steel Weld, Koskinen, Ari

- One scientific journal article, Producing a POD curve with emulated signal response data, Koskinen, Tuomas

- One Master's thesis completed on NDE tests and simulation of the cracks on austenitic steel weld, Koskinen, Tuomas

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. crystallinity degree, amount of fillers and antioxidants, has an effect to the ageing behaviour. Thus the degradation mechanism can be quite complex.

COMRADE is developed based on input from a feasibility studies from Energiforsk AB and STUK and through discussions between VTT, SP 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 2016

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 complete first set of accelerated ageing test on EPDM o-ring and from the test data obtained, to suggest preliminary acceptance criterion for this specific o-ring design. Also identification method of the vulcanizing agent from different EPDM blends was recognized. In WP2 the goal was to clarify whether polymer components aged at realistic conditions could be obtained for study from the closed down Barsebäck NPP. In WP3 various polymer ageing related factors are studied. First, related to the modelling work conducted with polymers, the goal was to recognize suitable computational modelling techniques that are applicable to model polymer ageing in different nuclear power plant applications. Second, the experimental work concentrated on studying synergistic effects of ionizing radiation and heat during DBA and evaluating different measurement techniques to measure oxidation profiles induced by ageing on sample surfaces. Third goal was to recognize models that could be used to predict severity of dose rate effect yielding from low dose rate irradiation.

During 2016 the first set of o-rings were aged and tested for their material properties at four different evaluation points and at one reference point for virgin material. The tightness failed for the first time for the o-ring running in 140°C (both irradiated and not irradiated) at evaluation 3. This is after five months of heat and four weeks of irradiation. The material properties at the end of life are listed in the Table 2.3.8.1 below. In evaluation 2 for the same temperature the function is still working even though material property for compression set being above 90% and elongation at break with a decrease of 41% thus showing severe material degradation. This indicates that very high material degradation is needed before the o-ring stops to function. The NMR indicates that there are differences between the irradiated a non-irradiated samples. It is difficult to see though studying the other material properties, see compression set as an example in the Figure 2.3.8.1.

Table 2.3.8.1. *Summary of material properties for the initial value and end of life when the functional property is tightness.*

Property	End of life	Initial value
Compression set	105%	4,9%
Hardness	80	72,3
Elongation at break	50%	182%
Tensile strength	7,5 MPa	12,8 MPa
DSC – OITe	235,9°C	265°C

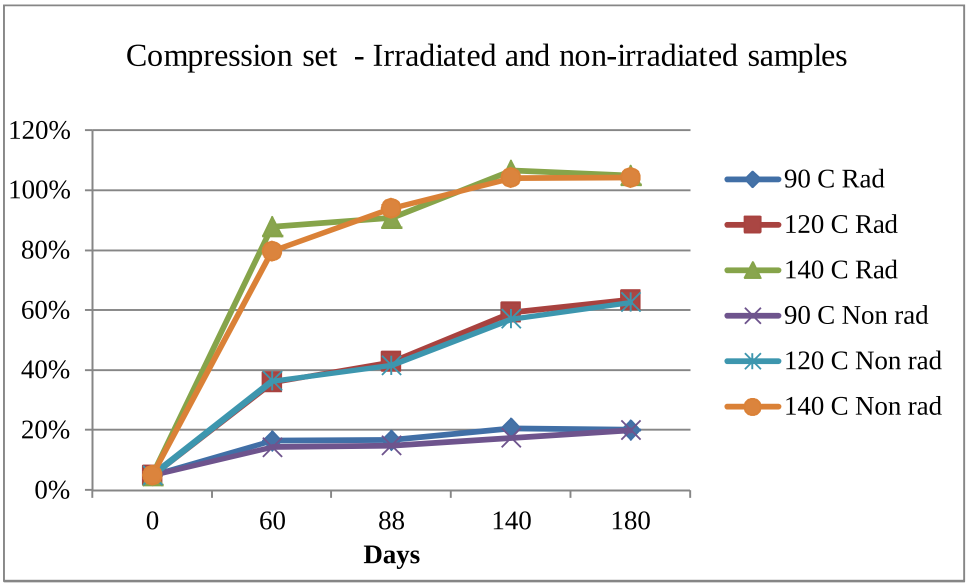


Figure 2.3.8.1. *Compression set measurement results at reference and four evaluation points for irradiated and non-irradiated samples aged at different temperatures.*

No experimental test has been done using material from a power plant as planned in the second goal. The pre study in WP2 will show what material is available and after that a small scale test can be started in WP1 to achieve the goal to validate the test method for the function of tightness. The third goal is planned for during 2017 and the beginning of 2018. Data gathered in the second ageing run using o-rings with cord diameter of 7 mm will provide needed data to be able to build the model using Finite Element Method (FEM). The fourth goal has not been planned for during 2016 and will be initiated during 2017. A master thesis student will join SP to work with this project and also with the deployment in the industry.

In WP2, two material experts related to Barsebäck polymer components were interviewed in order to clarify whether the used components could be taken out from the plant for ageing studies. The polymer components available for ageing study and which service conditions are well documented are few in numbers. Also radiological clearances and the related precautions to working with decontaminated materials yield in complicated and costly material acquisition. More feasible way to obtain used materials from properly documented service environments would be acquiring materials from running plants during annual take outs.

In WP3 modelling work related task literature survey was completed in 2016 on the synergistic effects of temperature and radiation in polymer ageing, and on possible ways of considering them in the lifetime prediction of nuclear power plant components. The following topics were looked into in more detail: the proposed mechanisms behind the synergistic effects, material modelling methods feasible for studying ageing and an overview of previous research related to the topic. The mechanisms underlying combined thermal and radiation ageing can be exceedingly complex, involving various chemical and physical processes across multiple structural and time scales. There remains a formidable gap between the present multi-scale materials modelling capabilities and practical lifetime prediction. Currently the most recognized practical lifetime prediction methods are semi-empirical and based on accelerated ageing experiments. Methods applicable to combined thermal and radiation ageing include the superposition of time-dependent data method, and the superposition of dose-to-equivalent-damage data method. The semi-empirical methods are limited in their predictive capability, as they cannot address possible changes in the dominant ageing mechanisms. For this reason, anomalous ageing phenomena such as the reverse temperature effect can render their predictions useless. In future the modelling work inside COMRADE focused on a particular synergistic mechanism or some other relevant detail of the ageing process is feasible, such as the reverse temperature effect.

The first goal in the experimental task was to study synergistic effects yielding from radiation and heat on EPDM and CSM rubbers. The samples were aged at three different temperatures and irradiated with three different absorbed doses. Based on the elongation at break results obtained with EPDM samples (Figure 2.3.8.2), it can be stated that moderate increase (ca. 75-125°C) in temperature during exposure to ionizing radiation hinders the degradation process. In addition, plane thermal ageing (equivalent to the thermal ageing component during simultaneous ageing) did not result in any changes in elongation at break. CSM (Lipalon) seemed to be more susceptible to both irradiation and thermally induced ageing and only small synergistic effects rising from simultaneous exposure to radiation and heat could be observed when simultaneous radiation and thermal ageing data was compared to plane thermal ageing data. Thermal ageing at 125°C resulted in clear decrease in elongation at break. Only slightly larger decrease was observed at 125°C when irradiation was conducted simultaneously. Simultaneous exposure to increasing temperature with irradiation resulted in increasing degradation. This behaviour was opposite to what was observed on EPDM samples.

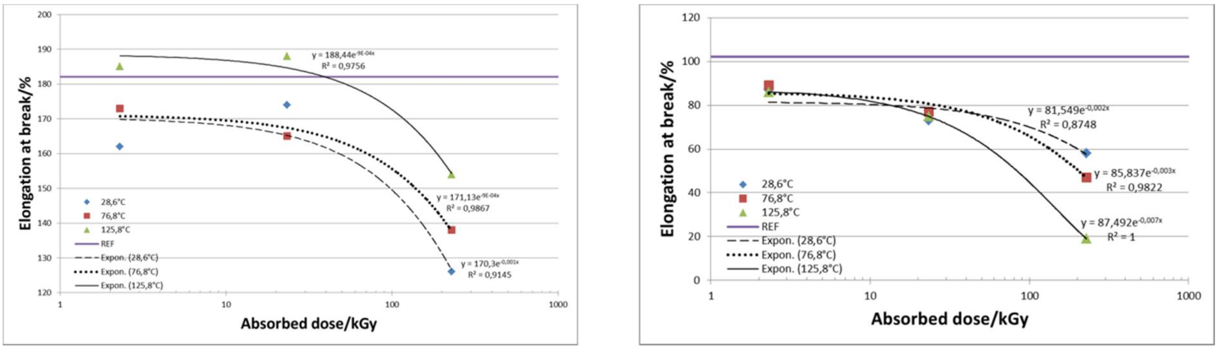


Figure 2.3.8.2. *Decrease of elongation at break as function of absorbed dose at different temperatures for EPDM (left) and CSM (right).*

The second goal in the experimental task was evaluating applicability of different techniques on measuring the oxidation profile created on EPDM samples during accelerated ageing and evaluate whether the measured oxidation profile could be correlated to mechanical properties of the sample material. Studied techniques included differential scanning calorimetry (DSC), time of flight secondary ion mass spectroscopy analysis (ToF SIMS) and Fourier transmission electron microscopy (FTIR). The studied material was sulphur and peroxide cured EPDM rubber and it was aged to three different conditions: thermally aged, gamma radiated and simultaneously (peroxide cured EPDM sequentially) thermally and gamma radiated. The overall condition of samples was evaluated by tensile testing (i.e. tensile strength, elongation at break and modulus at 100% strain). Based on the results obtained with ToF-SIMS, it can clearly detect the oxidation occurring in the vicinity of surfaces of the aged samples. However, careful sample preparation is required since the method is sensitive to surface roughness. The other two methods had limitations on the sample material (FTIR does not work on material containing carbon black) and resolution when creating an oxidation profile from surface towards the bulk (DSC). From the measured data obtained with these techniques, no correlation could be drawn between the mechanical properties of EPDM and the formed oxidation profile.

First annual COMRADE-project meeting was organized in Borås, Sweden during September 2016. Meeting had ca. 30 participants from research organisations, nuclear power plants, regulators and component manufacturers. Two day meeting was comprised of lecture day, where Dr. Burnay lectured on polymer ageing issues in nuclear power plant environments and of second day, where project results were presented and discussed and excursion to SP laboratories was organized. The meeting was considered to be successful and it is organized again in September 2017 at VTT.

Deliverables in 2016

- Development of condition monitoring technique for O-rings used in NPP applications (SP-research report).
- Identification of available polymers and their data from Barsebäck (SP-research report).
- Modelling tools for the combined effects of thermal and radiation ageing in polymeric materials (VTT-research report).
- Synergistic effects of radiation and heat on EPDM and CSM rubber (VTT-research report).

- Oxidation of EPDM: oxidation depth measurements and effects on material properties (VTT-research report).

- Methods used in evaluating the severity of dose rate effect. (VTT-research report).

2.4 Research infrastructure

In 2016 the research area “Research infrastructure” consisted of three projects:

1. Development of thermal-hydraulic infrastructure at LUT (INFRAL)
2. JHR collaboration & Melodie follow-up (JHR)
3. Radiological laboratory commissioning (RADLAB).

2.4.1 INFRAL - Development of thermal-hydraulic infrastructure at LUT

The aim of the INFRAL project is to develop the thermal hydraulic measurement infrastructure of the LUT (Lappeenranta University of Technology) 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 is the further development of the techniques related to the advanced measurements and their applications. The goal is 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 is 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 2016

In 2016, the INFRAL project was divided into four different work packages. The first work package (Advanced measurement techniques) includes activities that are related to the use of advanced measurement techniques at LUT. Part of the work is to develop analytical tools to extract the needed data from the measurements. The other part is to study the applicability of the techniques for different flow problems and to develop new measurement solutions. The second work package (Maintenance and equipment) aims on the maintenance of (PWR) PACTEL and other test facilities, and it comprises the yearly inspections, calibrations etc. The third work package (Modular Integral Test Facility (MOTEL)) aims 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) includes 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, are a part of the work package.

The work package 1 of INFRAL consists 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 on-going SAFIR2018 research programme, the advanced measurement systems have been developed further and used in multiple applications. In 2016, 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 systems are also related to non-SAFIR projects. The PIV measurement system, in particular, has been versatilely used in many different projects during 2016.

In 2015, the particle image velocimetry system was upgraded for shadowgraphy use with add-on components (software + hardware). Shadowgraphy is used to size water droplets with the help of bright backlight that creates a measurable edge of the particle's cross section. The end result is a droplet distribution from the measurement area, which can be used to determine the spread parameter for Rosin-Rammler distribution used commonly for droplet mass fraction frequency distributions. Figure 2.4.1.1 shows the workflow of a shadowgraphy analysis. The shadowgraphy extension was not utilized in 2016 for measurement activities, but the capability to execute new shadowgraphy measurements exists. A master's thesis “Spray Droplet Size Distribution Measurement” by Dmitry Skripnikov was finalized in September 2016. The thesis emphasized in handling of shadowgraphy droplet distribution data and ways to validate CFD (ANSYS Fluent) results with droplet mass fraction frequency distribution by providing correct spread diameter for Rosin-Rammler droplet number and volume distributions. The writing of the thesis was funded by LUT/Moscow Power Engineering Institute double degree program.

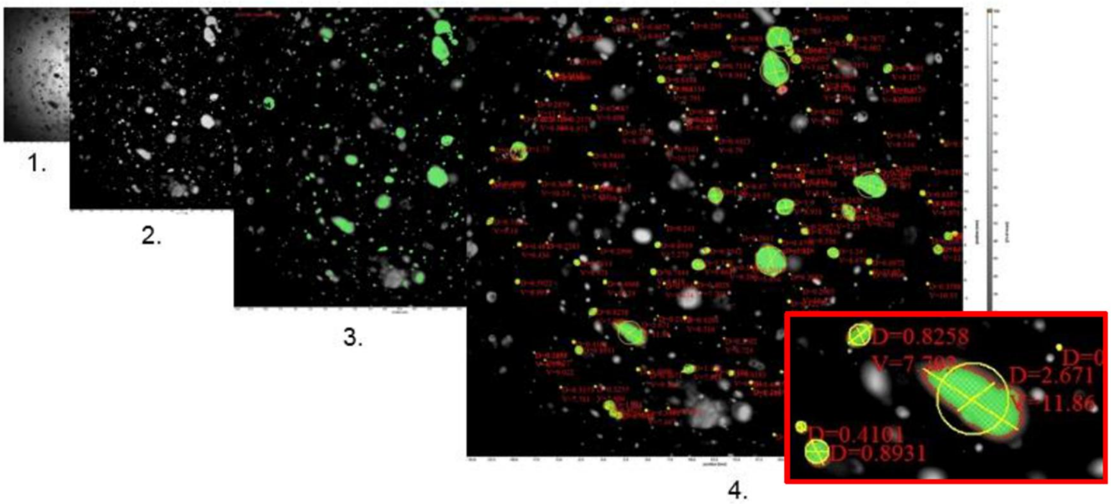


Figure 2.4.1.1. The workflow of droplet shadowgraphy measurements. Analyzation of the water droplets from the spray nozzle. 1. Raw image. 2. Processed image. 3. Droplet recognition. 4. Droplet properties (size, velocity, …).

In 2016, the PIV system was mainly utilized in contract measurement schemes, which have significantly broadened the PIV expertise. The most challenging scheme was the measurement of an 8 MW gas burner in a chamber. Measurement experience was also gathered from a simple airflow (stereo-PIV) and water flow within a channel (planar-PIV), which were more traditional measurements with more easily controlled and optically sturdier environments.

In December 2016, PIV was applied within the SPA-T8R and SPA-T9 test series where thermally separated layers of water were mixed with a sparger using different amounts of outlets open within the sparger. The test series was a part of the INSTAB project. As it has been found out before in case of condensation, the optical aberrations create challenging environment for successful PIV measurements. Another downside with the existing system is the low frequency of the laser and the cameras. Hence, they don't work well with the fluctuating water phase even without any optical aberrations, causing problems with time-averaging. The work to overcome these challenges technically still continues.

In December 2016, a new set of sCMOS cameras were acquired to update the PIV system. This update doubles the measurement frequency to 15 Hz (7 Hz before) making the laser unit the limiting component frequency-wise. The biggest advantage of the sCMOS cameras is the better chip design compared to the former ImagerX Pro cameras. The chip can withstand more reflection making it easier to apply in the challenging measurement environments. Another benefit of the new cameras is a better overall performance for shadowgraphy measurements.

The advanced applications for 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. In 2015, the applicability of the AXE sensor was studied under various flow conditions in the HIPE (Horizontal and Inclined Pipe Experiments) test facility. The results from the axial WMS measurements were presented in the SWINTH-2016 workshop in June 2016, where two researchers from LUT attended. Example of the results from the measurements is presented in Figure 2.4.1.2. The figure presents time-averaged void fraction distributions in the flow channel of the HIPE facility. These are measured with different pipe inclinations with both traditional radial, and axial wire-mesh sensors.

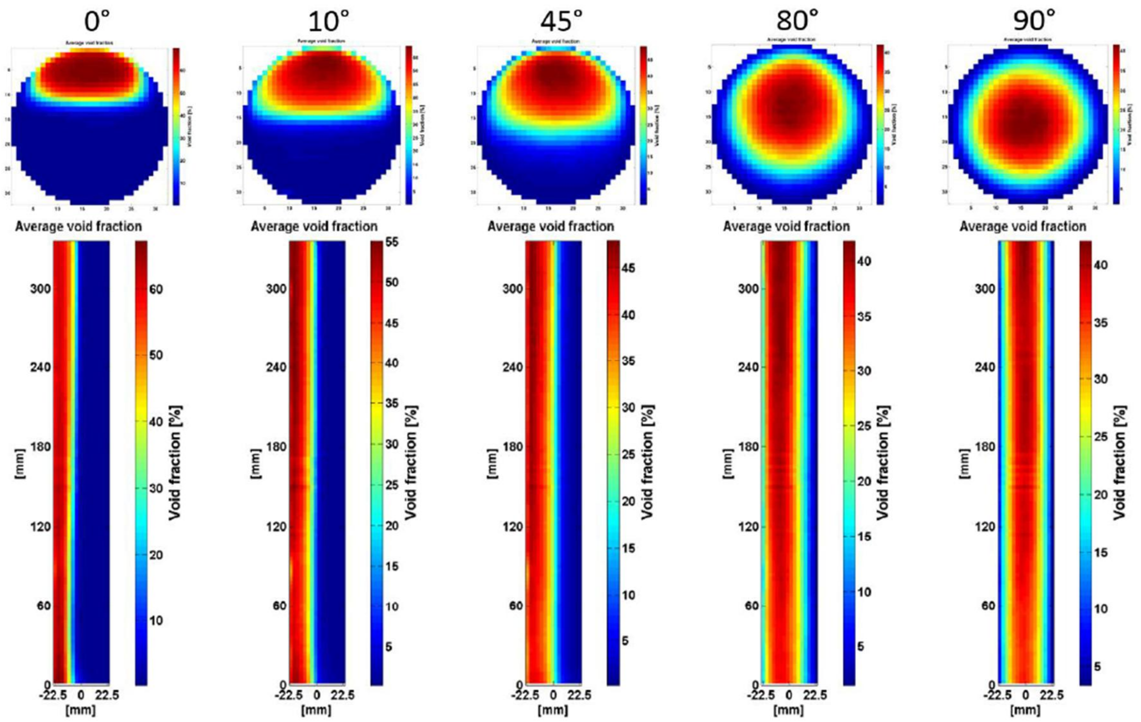


Figure 2.4.1.2. Time-averaged void fraction distributions (J_L=1.2 m/s, J_G=0.6 m/s), horizontal to vertical.

In 2016, the axial WMS technique was studied further. Two separate swirling devices with different blade angles (30 ° and 60 °) were designed and manufactured. The swirling devices were applied to the HIPE test facility to create a swirling two-phase flow. Two identical series of experiments were conducted separately with both swirling devices using both axial and radial sensors under swirling two-phase flow conditions. The data analysis of these experiments will be done later to study how the axial sensor models swirling two-phase flow.

In addition to developing own WMSs in the LUT laboratory, the development of the so-called high temperature and high pressure WMS technique at ETH Zurich and Helmholtz-Zentrum

Dresden-Rossendorf (HZDR) has been followed. The high temperature/high pressure WMS technique was also presented in the SWINTH-2016 workshop where good knowledge of it was acquired. The sensors can operate at temperatures potentially up to 350 °C and pressures up to 22 MPa. There have been preliminary plans of applying this technique to the forthcoming modular integral test facility, MOTEL, which is under design at LUT. The temperature and pressure levels will be so high in the test facility that the traditional WMSs cannot be applied to it. High temperature/high pressure WMSs could offer a novel means of void fraction measuring in challenging circumstances.

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 develop for analyzation of the HSC measurement data. In 2016, only analysis of the old experiments were made, and there were no new measurements with high-speed cameras concerning pattern recognition. During 2016, the pattern recognition algorithm made for the DCC-05 experiments was improved. It is now possible to calculate volume, surface area, diameter, condensation velocity/acceleration, and (chugging) frequency of condensing bubbles. The preliminary results were published in Nuclear Engineering and Design ("Direct contact condensation modeling in pressure suppression pool system"). The article is linked also to the NURESA and INSTAB projects. More detailed analysis will be presented in another journal article, as well as in the NURETH-17 conference in 2017. The analysis for previous PPOOLEX experiments with a transparent blowdown pipe was started during fall 2016.

In addition to PIV, WMSs and HSCs, development of other advanced measurement techniques is 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 enables the measurement of temperature distribution in high detail in different geometries, such as a slab or a rod. There have been preliminary plans for utilizing distributed temperature sensors in the LUT laboratory in the future. Different applications of the DTS technique are presented in Figure 2.4.1.3.

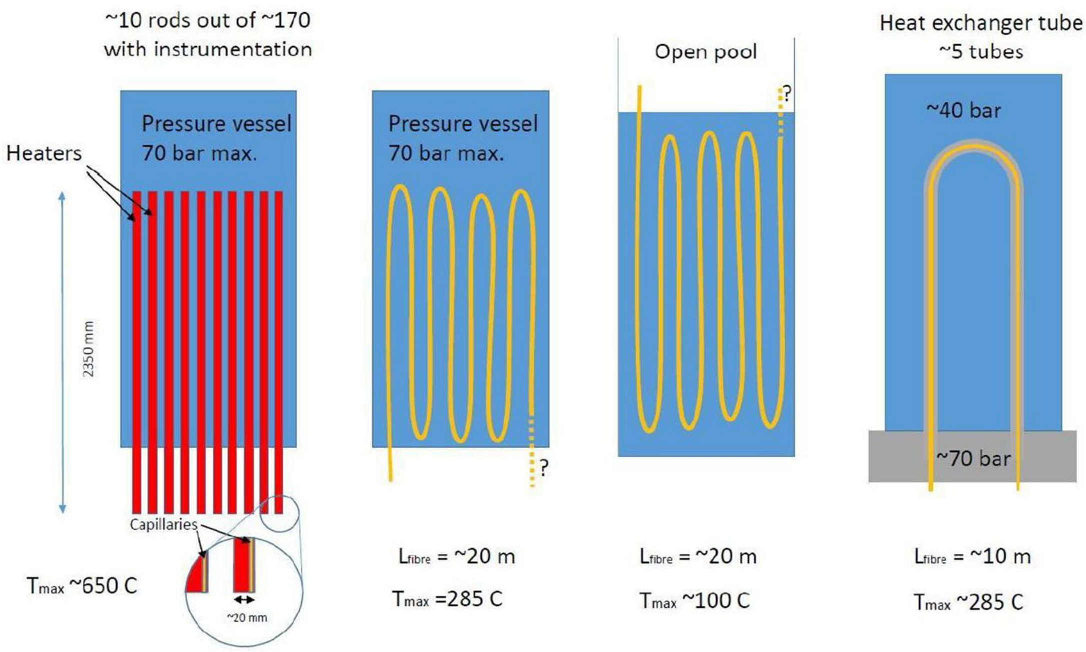


Figure 2.4.1.3. Different applications of the distributed temperature sensor.

In addition to the DTS technique, the development of different tomography measurement systems has been followed. The research visit to University of Michigan (UMICH), which took place in September–October 2016, provided lots of important information concerning tomography systems and their characteristics, such as technical and economic requirements. Two researchers from LUT attended the research visit, and during the visit the tomography measurement systems of UMICH Nuclear Engineering and Radiological Sciences department were introduced. The visit provided also valuable experience and information on e.g. Laser Doppler Velocimetry (LDV) measurements and WMS data processing procedures.

Within the work package 2, the yearly calibrations of the (PWR) PACTEL measurements were carried out during the summer without serious problems. Some of the components, such as pressure and pressure difference gauges, have been renewed. The periodical pressure vessel inspections were successfully conducted. The power measurement system of PACTEL was renewed. During 2016, a leakage in the PACTEL pressurizer relief valve was noticed. The valve seat was lapped and new gaskets fitted, but seemingly the leakage has continued. The issue is being followed. In 2016, the upgrade of power transformers was planned to enable higher heating power to be available for the experiments (1 MW → appr. 2.3 MW). The upgrade process has been postponed, and the options for the upgrade are being studied.

In the work package 3, the survey of the research based requirements for the forthcoming modular integral test facility (MOTEL) was carried out. A research report was written on the issue. The report introduces the international trends and future needs in (experimental) thermal hydraulic research, and the research based requirements and guidelines for the design of MOTEL. The report is based on recent international articles on global experimental needs and challenges, as well as on domestic research needs in Finland. The design and the construction of the first components of the facility will be carried out during 2016–2018, and the experimental activities will begin in 2019. The actual design of MOTEL started in 2016 with a survey of the options for the first heater element. The design and the construction are funded by the Academy of Finland.

Deliverables in 2016

- Status report on the advances in thermal-hydraulic measurements (WP1 report).
- Participation to the SWINTH-2016 workshop (Livorno, Italy, June 2016). The workshop paper is titled “Estimation of velocity fields from the axial wire-mesh sensor data”.
- Maintenance of PWR PACTEL / PACTEL.
- Research based requirements of “MOTEL” -report. Report describes the main research based requirements for the forthcoming modular integral test facility.
- Research visit to University of Michigan. A travel report was written.
- International co-operation activities.

2.4.2 JHR - JHR collaboration & Melodie follow-up

The general objective of the SAFIR JHR project was to follow the progress of the Jules Horowitz Reactor (JHR) international project by participating in the work of three Working Groups (WG), namely Fuel WG (FWG), Materials WG (MWG), and Technology WG (TWG), established within the JHR consortium. Furthermore, participation in the performance of the Melodie in-core experiment in collaboration with CEA was one of the main goals.

The objectives of the WGs are the determination of experimental needs, the planning of future experiments, and the development of experimental devices and infrastructure. Some of the experimental devices are based on existing technologies, but also new types of devices are being developed, extending the experimental capabilities and bringing new information on the subjects studied. The Finnish in-kind contribution to JHR gives an access to these technologies and enables international collaboration in the future experiments.

The participation in the three WGs brings knowledge on nuclear fuel and irradiated materials research as well as on the preparation and execution of in-core experiments to Finland, and this knowledge will be disseminated to the SAFIR2018 community. Through the participation in the working groups, it is possible to bring forward our national interests with regard to nuclear materials research.

The second work package of this project focuses on following the Melodie experiment and bringing the knowledge on the feasibility of the technology as well as the data back to VTT and the SAFIR2018 community. The delivery of the Melodie device was part of the Finnish in-kind contribution, and in this part of the work was successfully completed in 2012. The Melodie in-core experiment, carried out in the Osiris reactor at CEA Saclay research centre in 2015, aimed at validating the use of the device and its novel technology for future experiments in JHR.

Specific goals in 2016

In 2016, the project members participated in two WG meetings as well as the annual JHR technical seminar, which was combined with the Nugenia Forum. The goal was to participate in the planning work of the WGs and enable a bidirectional exchange of information. Following the work carried out by the WGs in 2015, one of the main goals was to start a detailed planning of the first pre-JHR experiments.

The first meeting of all three WGs was organised in Israel. In this meeting, the two documents to be delivered to the governing board were reviewed once more. These two documents were the synthesis document describing the work carried out by the WGs so far and the objectives for the future, and the position paper describing a proposal for two irradiation experiments to be carried out before the start-up of JHR. Furthermore, the initial planning of a Euratom H2020 project proposal called FIJHOP was launched.

The sixth annual JHR Technical Seminar was combined with the Nugenia Forum, which was held in Marseille in April 2016. The main goal in combining the technical seminar with the Nugenia Forum was to give an update on the JHR construction project, describe the work carried out by the WGs and to introduce the FIJHOP project proposal to the Nugenia community to attract interest among the participants, especially those not part of the JHR consortium. Furthermore, the initial results of the Melodie in-core experiment were presented as an example of innovative, highly instrumented irradiation devices envisioned for future experiments in JHR.

A national JHR seminar was held in the VTT Centre for Nuclear Safety in October 2016. The goal of this seminar was to share information on the progress of the JHR project to the Finnish stakeholders.

Another WG meeting took place in Mol, Belgium in November 2016, again with all three WGs together. In this meeting, the WGs discussed the technical specifications of the irradiation experiments planned to be carried out in the FIJHOP project.

In the Melodie follow-up, the goal was to collect data produced in the in-core experiment carried out in 2015. Furthermore, the goal was to gather information on data analysis, including both the data from the sample holder instrumentation and the data from the Chouca capsule and Osiris instrumentation. These goals were abandoned during 2016 because of the slow progress of the disassembly of the Melodie sample holder and the workload of the

WG participation, which was higher than expected. Resources from the Melodie follow-up were reallocated to the WG participation.

Deliverables in 2016

- Project members participated in the WG meetings
- Travel reports were written for both WG meetings
- Proposal for the FIJHOP project including the pre-JHR irradiation experiments was successfully submitted
- National JHR seminar was organised in October 2016

2.4.3 RADLAB – Radiological laboratory commissioning

The safe research and testing of radioactive and contaminated materials of the nuclear sector requires radiological facilities, and for highly radioactive materials, full hot cells for remotely handling the materials inside of heavy gamma shielding. The objective of the RADLAB project was to plan and execute the hot cell and hot laboratory portion of the radiological research infrastructure renewal, including the planning and making of critical equipment investments for the facility, and training of the technical personnel that will be staffing the facility, carried out in tandem with the completion and commissioning of the new VTT Centre for Nuclear Safety (CNS). The project was executed in 5 main work packages: the first one focused on the hot cell design, fabrication and commissioning process; the second one tasked with hot laboratory equipment procurement; the third one dedicated to self-built research facilities; the fourth one aimed at self-built facilities for handling and storage of radioactive materials and waste; and finally, in light of the integral nature of this work with the realization of the VTT CNS, a fifth work package focused mainly on the organization and ramp-up of the VTT CNS.

Specific goals in 2016

The design and construction of the hot laboratory facility involves defining and guiding the technical aspects of the hot laboratory portion of the new laboratory in tandem with the realization of the Centre for Nuclear Safety (CNS) building itself. Central to the 2016 project year was fabrication of the hot cells, carried out as a subcontract with Isotope Technologies Dresden, GmbH, based on the design work carried out with them in 2015.

Development of remote operation methods and skills is important for effective utilization of the new infrastructure. In 2016 these methods were developed through an extended visit by a hot cell engineer to Paul Scherrer Institute (PSI) in Switzerland.

To ensure that equipment can be operated in a radiation environment and operated remotely, “nuclearization” can be necessary. The work on nuclearization and in-cell devices mainly focused in 2016 on assessing robotic versus CNC remote machining operations for opening surveillance capsules, on evaluation of different semi-automatic dimensioning microscope candidates, and further assessment of hot-cell pre-fatiguing equipment candidates.

Procurement of research equipment for installation in the hot laboratory facilities is an important area of effort in the infrastructure renewal. In 2016 several equipment investments were scheduled to coincide with the manufacturing of the hot cells and the completion of the laboratory facilities. The principle pieces of equipment for purchase in 2016 were an impact test hammer with semi-automatic specimen feeding, and a universal mechanical testing system (UTS) with an environmental chamber, both for integration into the hot cells. Delivery of a heavy-duty pallet truck for handling large transport casks was scheduled to allow demonstrating its functionality during the hot cell factory acceptance test. Additionally, slated for purchase and installation directly in the new building as the laboratory wing was completed, were the radiation monitoring systems for the CNS, as well as devices for the mechanical support workshop. Finally, the procurement of a Talos transmission electron microscope that was prepared in 2015, was scheduled to conclude in 2016 by purchase and installation of the device directly in the new laboratories, along with installation of the Zeiss Cross-Beam scanning electron microscope that was purchased and delivered in 2015.

Some research and testing devices are not readily available on the market, but rather, require custom design. Development and construction of those research devices is carried out with the experts involved in utilizing the equipment for producing research results, and are then fabricated by in-house assembly of parts bought from component suppliers, or made by in-house or outside workshops. An important goal in 2016 was the procurement of components for refurbishing the hot autoclaves and water circuit, to be located in a dedicated room of the basement of the CNS. Another goal was the design of localized shielding for some large mechanical testing devices, to enable construction during 2017. Finally, device nuclearization needs were identified for equipment installation into the hot cells, including modifications to the electron beam welder and electric-discharge machining device.

The VTT CNS requires a number of supporting facilities for its research and testing operations. It was determined that the facilities for handling and storage of radioactive materials and waste could be more cost effective for VTT to design, fabricate and install themselves, rather than to try to include them as a part of the main hot cell suite contract. The self-built supporting facilities task in 2016 continued with the development of the facilities for three main areas: laboratory radioactive waste handling, radioactive research material logistics, and orderly temporary storage of radioactive specimens. These systems are mainly to be located in the basement of the CNS.

In 2016 the construction of the VTT CNS concluded, and therefore a principal goal was the move-in of equipment and the radiological commissioning of the laboratory.

Deliverables in the RADLAB Hot Cells work package in 2016

This work package, with a focus on the hot cell manufacturing and installation, and training for remote handling work, has progressed steadily in 2016. Some slight delays accumulated to affect the Factory Acceptance Test (FAT) schedule at Isotope Technologies Dresden (ITD), but otherwise the installation work is on schedule.

- The detailed design and manufacturing drawings for the hot cells were completed early in the year. By the end of the year, the fabrication of the hot cells at ITD was completed for most of the units. All the factory acceptances tests were also completed, except that of the microscopy cell unit.

- Installation of the hot cells in VTT's facilities commenced on the first day of business of 2017. The challenging first set is already taking shape (see Figure 2.4.3.1). That includes the heavy utility cell in the basement, the vertical elevator connecting the basement cell with the horizontal transport rail, and the two anchor cells of the two rows of main suite cells containing the terminuses of the transport rail.

- In quality management, after a few iterations between VTT and ITD (assisted by Qualifinn expertise), a satisfactory Manufacturing Quality Plan was completed by ITD, and the project deliverable on the topic was completed.

Figure 2.4.3.1: Installation of the hot cells in the VTT Centre for Nuclear Safety began at the beginning of 2017.

- On the technical training side, a hot cell engineer completed a three months stay at PSI. During the stay, he became more familiar with remote preparation and handling of metallography specimens for analyses by electron probe microanalysis (EPMA) and secondary ion mass spectrometry (SIMS), remote execution of mechanical tests, and utilization of robotic manipulation in a hot cell setting. The microstructural characterization results that he gained as a side benefit of his stay at PSI were incorporated into his Master's Thesis, which was awarded a scholarship from the Yrjö ja Senja Koivusen säätiön.

Deliverables in the RADLAB Equipment Procurement work package in 2016

This WP, focused on executing the procurement process for series-produced equipment for the hot laboratory, progressed mainly as expected. Besides management of the entire multi-year investment portfolio, some specific devices were featured for delivery in this project year.

- Pre-fatiguing is an essential and time-consuming phase of fracture mechanical testing of RPV surveillance specimens. For a hot-cell ready device for pre-fatiguing, the technical description and offer from SCK-CEN for a device for cassette-loaded 10x10 Charpy-size test bars were compared with the technical description of a "commercial" device that could also pre-fatigue compact-tension specimens. Specification papers were then written up in preparation for executing the tender in 2017.

- RPV surveillance capsules are typically comprised of a stainless-steel sheet-metal "can" containing pre-machined fracture toughness specimens. The first step upon receiving a surveillance capsule for testing, is to machine open the capsule and recover the specimens. Since the total radioactivity can be quite high, this must be done remotely. In 2016 a report describing the trials with in-cell robot-assisted machining for this task was compiled and delivered. The trials showed that a robot-mounted saw can be used to cut the metal of a surveillance capsule, and then a manipulator jaw can be utilized by the robot to pick up individual specimens for closer visual inspection of e.g. identity stamps.

- An alternative approach to opening surveillance capsules is by more conventional CNC machining. After discussions with several different English and Finnish suppliers of custom CNC devices, in 2016 a 3-axis mill with remote CNC capabilities was conceptually designed for VTT by Metecno Oy. They also provided an initial cost estimate to manufacture one. That design was then iterated to an improved design that would be sufficient for making a purchase decision. Purchase of the device is now on hold while efforts focus on getting the hot cells up and running.

- Fracture toughness testing of RPV surveillance specimens is to be done primarily within the hot-cells. In 2016 the details of the in-cell mechanical test devices were finally fixed, including the details of the environment chamber for the universal testing machine. Then the final package price was negotiated with Zwick. The purchase orders for both mechanical test devices were approved in May, and the factory acceptance tests of both devices were carried out in November 2016 at Zwick's German factories. From the factories the devices were delivered to ITD for integration into the hot-cells. The FAT of the cells with integrated mechanical testing devices took place in early 2017.

- For observing the effects of neutron irradiation on material microstructure and microchemistry distribution, analytical transmission electron microscopy (TEM) is an essential tool. In 2015 the procurement process was launched for a FEI Talos F200X FEG TEM, with its built-in 4-sector energy dispersive x-ray spectrometer (EDS) and electron energy loss spectrometry (EELS) system. In 2016 the device was ultimately purchased as a VTT investment, and was successfully installed in the new CNS in August. User training on the TEM was carried out in November (not funded by RADLAB).

- For conducting failure analyses of components from NPPs, an essential tool is analytical scanning electron microscopy (SEM). In 2015 a Zeiss Crossbeam 540 SEM was procured, along with an EDAX "Triade" analyzer set-up containing three detectors (electron back-scatter detection, and both wavelength and energy dispersive x-ray spectrometers). The device is designed to be equipped with a focused-ion beam (FIB) accessory, which can be used as a precise "micro-knife" to excise material from specific locations for subsequent examination as a SEM cross-section, or by TEM. In 2016 the microscope was moved to its final location in the CNS, after which several users were given more training by the device manufacturer.

Deliverables in the RADLAB Research Equipment work package in 2016

- The hot-autoclave facilities enable testing primary circuit materials in simulated LWR water chemistry and temperatures. In 2016 the realization of the autoclave testing facilities progressed such that refurbishment components were procured and the refurbishment was carried out for one of the autoclaves. The second autoclave was found to have a crack though, and therefore must be replaced in 2017. For one of the water circuits, it was decided that an existing compact one can be moved to the CNS, in exchange for a larger one being decontaminated and then taken into use for non-active materials in VTT's underground research hall. A second water circuit can then be purchased later if needed.

- Locally-shielded equipment installations enable reference tests and tests of contaminated or low activity materials to be safely and easily executed outside the main hot cells, increasing testing capacity. To accommodate such installations, numerous exhaust ducts were incorporated in the main high bay, which can be connected to local containments. In addition to mechanical testing of metallic specimens, the requirements for testing irradiated concrete were considered. In 2016 the local shielding conceptual design was carried out for local shielding of two principal devices in the A-class high bay: an existing MTS tensile/pre-fatigue device, and a new full-sized Zwick impact hammer.

An existing Instron tensile device is accommodated by the MTS shielding. The 3-D models of the testing machines were made and then integrated with 3-D models of shielding concepts. Those were then iterated in collaboration with the mechanical testing experts (users), to come up with a final design for realization in the 2017 project.

- The electric discharge machine (EDM) is a key device for the cutting of radioactive materials. For example, accurate test specimen fabrication for surveillance programs requires a good EDM machine. The small amount of waste and a flexible cutting process are the most important features of the device. However, the device uses a closed water circuit, and the water must be very clean. As a part of the nuclearization of the device, a pilot set-up for removing the radioactive cutting debris from the EDM water circuit was built and tested over several years, to identify operating parameters for optimal functionality. In 2016, a report describing the EDM nuclearization and water circuit was completed and delivered to the project, including process component selection, operating parameter recommendations, and estimates of radioactivity exposure in different waste handling scenarios.

- The electron beam welder (EBW) is an essential device for reconstituting tested fracture toughness specimen pieces into new test specimens, which in turn enables more data generation from the same volume of irradiated test material. As a part of the hot cell design process in 2016, the nuclearization needs of the EBW were identified, and discussions launched with the EBW supplier. A series of adaptations and their cost were agreed with the EBW supplier, and purchase of the nuclearization changes was approved by the Steering Group as an investment. The required work will be carried out in conjunction with the installation of the EBW device into the hot cell in March 2017.

Deliverables in the RADLAB Supporting Facilities work package in 2016

- Proper sorting, consolidation, packaging and temporary storing of radioactive waste is essential for the day-to-day operations of the new radiological laboratory. Therefore, effort has gone to designing and realizing waste handling infrastructure. Once the new building was turned over to VTT in 2016, the facility installation conditions were assessed and a few changes were requested to achieve better functionality for the waste handling rooms (e.g. lighting raised, electrical box and radiator locations moved).

- After a procurement process, Platom Oy was awarded the subcontract for the detailed designing of a wet-waste handling facility. Delays in the procurement process meant that the contract did not get underway until the last couple of weeks of 2016, and the completion is expected in March 2017. But within the 2016 project period, a draft concept of a liquid waste evaporation system in a glove box was produced.

- For handling the waste storage barrels in the temporary storage room, a light-duty bridge crane was specified. A tendering process for its procurement concluded in 2016 by awarding it to Erikkila Oy. To tailor the crane for improved radiation safety, the company is incorporating a remote operation capability, and redesigning the barrel clamp to ensure a safe grip on the barrels. It will be installed in the CNS in the first half of 2017.

- For handling the heavy transport casks containing radioactive materials for research and testing, a heavy duty pallet truck was ordered from Genkinger-HUBTEX GmbH. It was first delivered to ITD and used in the FAT of hot cell group #3 in October, and then delivered to VTT along with the first hot cell components in November.

- The orderly storage of radioactive research materials is important for ensuring that the radioactive inventory is known, and that particular specimens can be positively identified and associated with specific relevant information. For that reason, in 2015 a conceptual design was carried out with Fraktio Oy for an electronic database system for the radioactive materials of the CNS. In 2016 a tendering process for the realization of the specimen database system awarded the contract to Ambientia Oy. The realization work began in June with a survey of end-user wishes by Ambientia, and coding work began in earnest after the summer holidays. The first version of the new software tool was delivered to VTT at the end of December 2016 for testing by VTT users. The final version of the software, dubbed Pergament, was delivered at the end of January 2017. It not only enables recording of specimen locations, but can retain the history of their movements through the testing processes in the facility. Specimens are recorded in the database together with basic necessary technical data and customer association, and the test result documentation can be linked to it.

- For the orderly physical storage of radioactive materials, two concepts were made. Each was comprised of a series of alphanumerically organized compartments, located within a gamma-shielding basin. One design consisted of long, narrow, horizontal drawers with a series of compartments holding small, removable specimen "baskets." The other was had vertical stacks of such compartments with removable baskets. After internal evaluation, the vertical stack version was selected for further design iteration. It would have 20% more capacity than the current storage facility slated for decommissioning, but like that one, it will have 3 different sizes of lockers, including some with a 5 x 5 grid of slots for e.g. Charpy specimens. That design was then evolved during 2016 to produce a design specification for the tendering process for the detailed design and manufacturing of the device in 2017.

Deliverables in the RADLAB VTT CNS work package in 2016

- The laboratory wing of the CNS was released to VTT on schedule in May 2016 (Figure 2.4.3.2). Many small fixes and changes were still carried out intensively in the weeks following. Some defects in the epoxy floor of the high-bay were revealed later in the summer, so the epoxy floor was renewed in October-November. Then over the Christmas break, a water leak in one room revealed some significant deficiencies in the capacity of the room to retain water, which was an important design criteria related to the Hi-Fog fire suppression system. Water-damage repairs are expected to be completed by the end of March 2017, and pre-emptive repairs will be made where necessary to ensure that water stays contained within rooms in the future.

- The radiological commissioning process of the new laboratory was managed through a few meetings with STUK over the 2016 period, and the documentation was prepared and submitted for three different permits. In late January 2017, STUK approved the radiological facility permit enabling operation of the CNS radiochemistry and microscopy laboratories for radioactive materials analyses. At the same time the nuclear materials security permit granted in the autumn of 2016 came into effect. The iodine laboratory of VTT Expert Services received their complementing permit in autumn 2016 as well.

- With the permits coming on line, it was a suitable time to hold a seminar detailing the new infrastructure capabilities more broadly. This Users' Group seminar was held on January 24th, 2017, and was followed by a tour of the new laboratories and hot-cell installation site.

- Besides the direct technical work involved in the infrastructure renewal that is a part of the RADLAB project, activities in 2016 focused on launching the VTT CNS as a national infrastructure with high demand for its services:
    - VTT and Fortum had a joint stand at the World Nuclear Expo in June 2016, which featured the CNS capabilities.

- An official inauguration event was held on September 20th, 2016 with a focus on Finnish clientele.
- New marketing materials were developed to profile the CNS nuclear offerings.
- VTT's web-pages were updated with new marketing materials about the CNS.
- A nuclear research display was prepared for the CNS lobby, profiling Finnish nuclear competencies over the decades.
- A new nuclear services marketing video featuring the CNS was made.

- In its first half year of existence, over 20 different tours and presentations were made of the new laboratory facility, with guests from both Finnish as well as international clientele.
- In 2016, write-ups of the new facility were featured in the Finnish Nuclear Society magazine ATS Ydintekniikka, the ETSON EUROSAFE magazine, Senaatikiinteistö's Kontrahti -magazine, Energiauutiset, many of the local Finnish papers affiliated with Lännenmedia, as well as a more technical presentation in the ATS commemoration symposium.

Figure 2.4.3.2: The construction of the VTT Centre for Nuclear Safety laboratory wing was completed in May 2016. The office wing was taken into use already in February 2016.

3. Financial and statistical information

The planned and realised volumes of the SAFIR2018 programme in 2016 were 6,83 M€ and 7,02 M€ and 44 and 50 person years, respectively. The funding partners were VYR with 4,073 M€, VTT with 1,460 M€, Lappeenranta University of Technology with 0,224 M€, Aalto University with 0,200 M€, SSM with 0,164 M€, NKS with 0,150 M€, Halden Reactor Project with 0,131 M€, and other partners with 0,620 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.

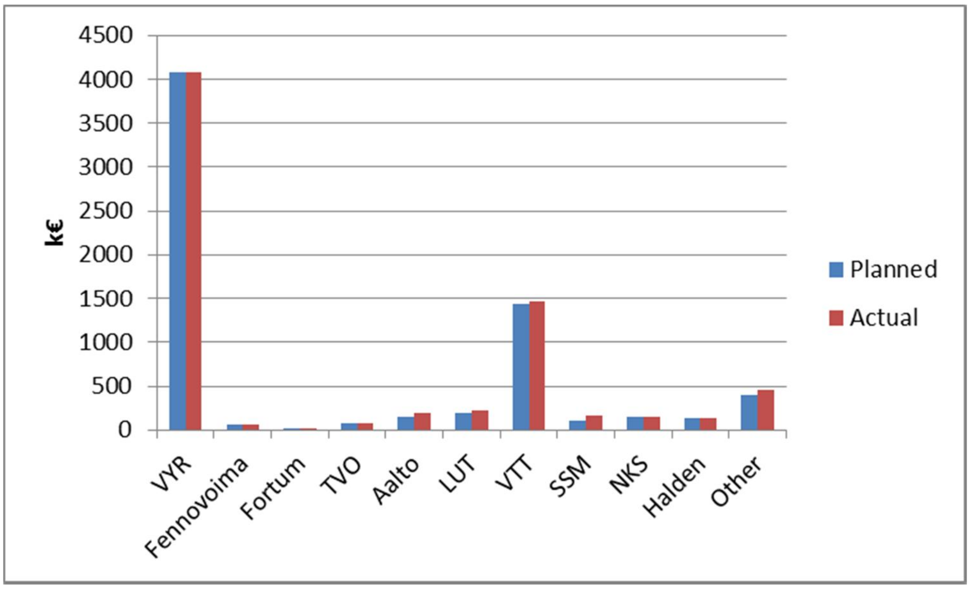


Figure 3.1. Planned and actual funding of the SAFIR2018 programme in 2016.

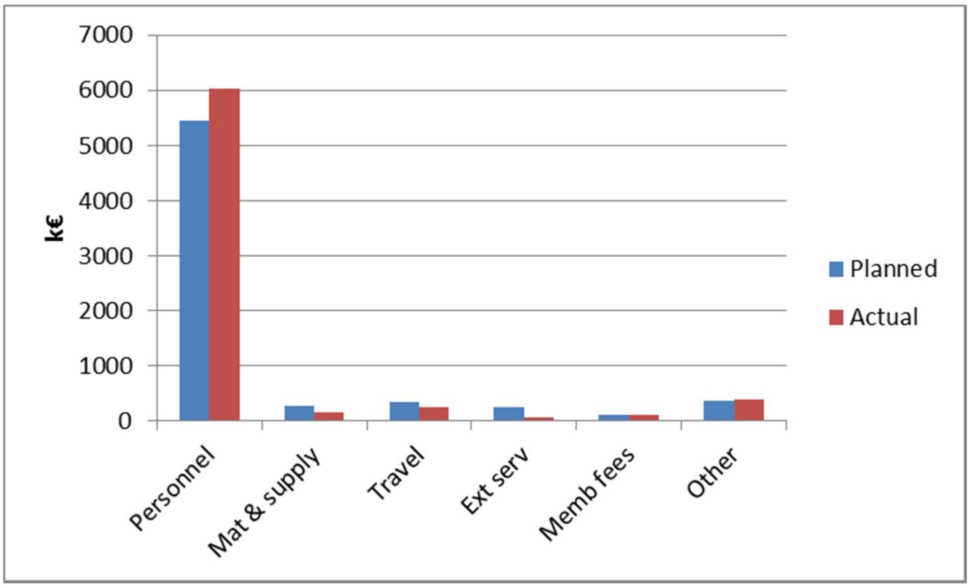


Figure 3.2. Planned and actual costs of the SAFIR2018 programme in 2016.

Figures 3.3-3.6 show the cost and volume distributions 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.

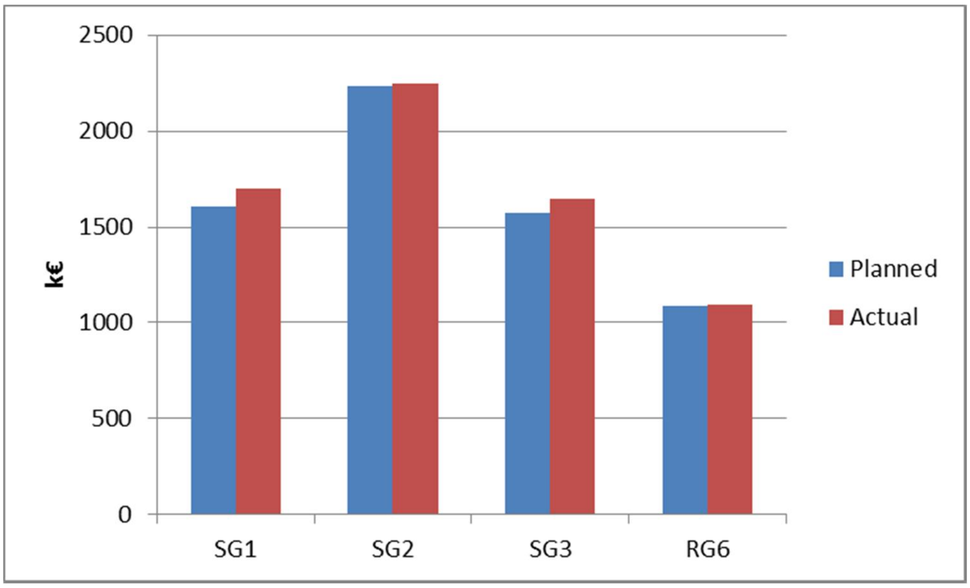


Figure 3.3. Planned and actual costs by research area in 2016.

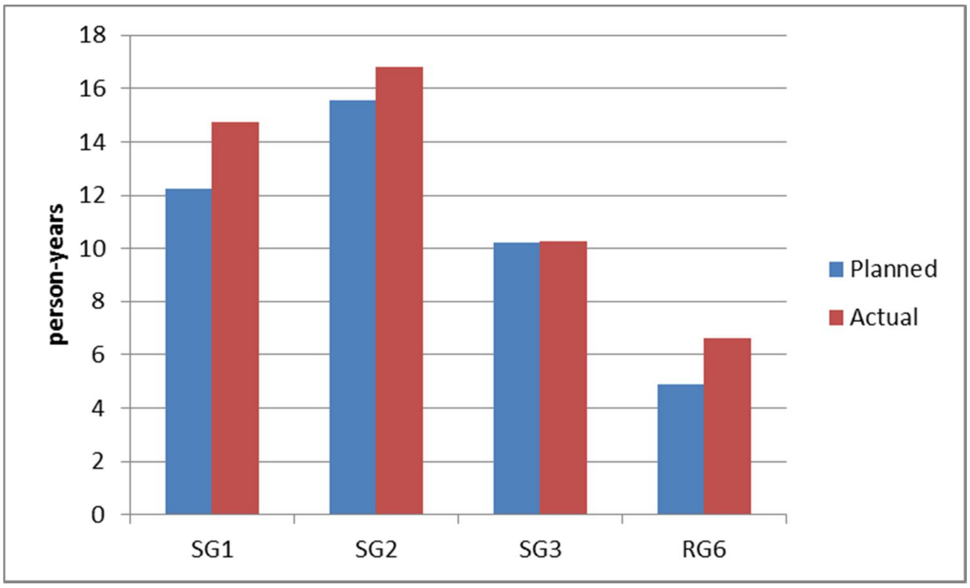


Figure 3.4. Planned and actual volumes by research area in 2016.

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 area Research infrastructure (RG6) the share of person-years was lower than the share of total funding because of infrastructure investments and subcontracting (Figures 3.5-3.6). In SG1 and SG2 the shares of the person-years were bigger and in SG3 smaller than the shares of the total funding, respectively.

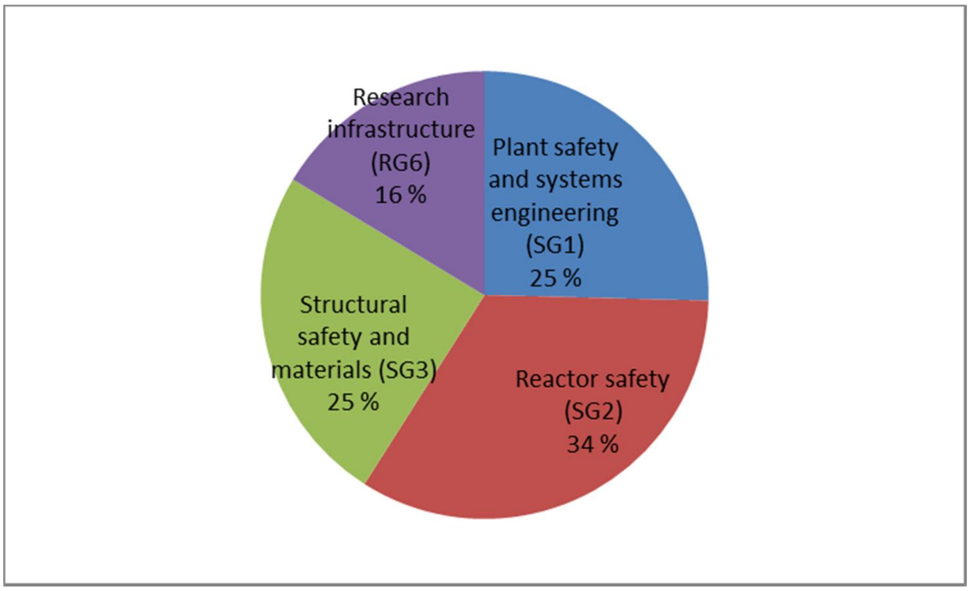


Figure 3.5. Distribution of total funding in SAFIR2018 research areas in 2016.

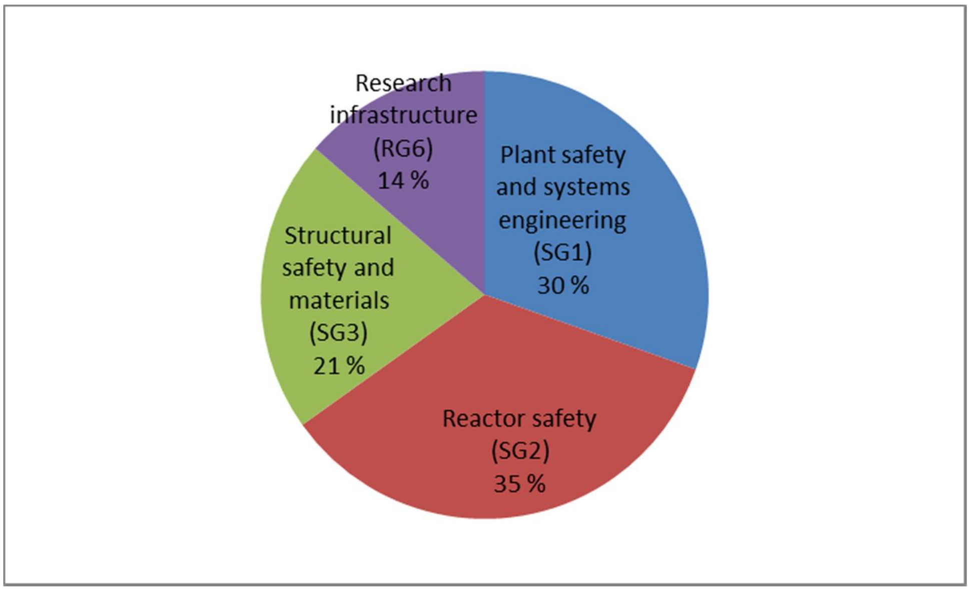


Figure 3.6. Distribution of person-years in SAFIR2018 research areas in 2016.

The numbers of different kind of publications made in SAFIR2018 research projects during 2016 are listed in Table 3.1. The programme produced 265 publications in 2016 consisting of 40 scientific journal articles, 50 conference articles, 104 research reports of the participating organisations, and 71 other publications (theses, reports of other organisations, etc.).

The average number of publications in the research projects was 5,3 per person-year, and the average number of scientific journal articles was 0,8 per person-year. There were clear differences in the number and type of publications between the projects. Some projects focused more on writing research reports while less scientific and conference articles were written in 2016.

Table 3.1. Publications in the SAFIR2018 projects in 2016.

Project acronym	Volume (person years)	Research reports	Scientific journal articles	Conference articles	Others	**Total** (number of publications)
CORE	2,0	1	1	10	3	15
EXWE	3,8	9	5	0	9	23
MAPS	1,6	0	2	2	9	13
PRAMEA	2,7	5	4	6	4	19
SAUNA	4,0	5	2	10	6	23
GENXFIN	0,6	2	1	0	4	7
CASA	1,5	5	1	0	6	12
CATFIS	0,9	3	4	3	2	12
COVA	1,8	8	0	0	0	8
INSTAB	1,6	4	1	0	0	5
INTEGRA	2,9	2	0	0	1	3
KATVE	1,3	5	0	3	2	10
MONSOON	1,0	0	3	0	3	6
NURESA	1,5	7	1	0	0	8
PANCHO	2,1	5	4	1	1	11
SADE	0,8	0	3	2	3	8
USVA	1,3	3	1	1	2	7
ERNEST	0,5	2	0	2	0	4
FIRED	1,5	2	1	0	1	4
FOUND	2,7	13	0	1	1	15
LOST	1,5	7	2	1	0	10
MOCCA	0,8	3	1	0	0	4
THELMA	1,8	2	2	1	4	9
WANDA	0,6	1	1	7	1	10
COMRADE	0,8	6	0	0	0	6
INFRAL	1,9	1	0	0	2	3
JHR	0,2	0	0	0	1	1
RADLAB	4,5	0	0	0	5	5
ADMIRE	1,0	3	0	0	1	4
Total	**49,5**	**104**	**40**	**50**	**71**	**265**

Altogether 13 higher academic degrees were obtained in the research projects in 2016: two Doctoral degrees and eleven Master's degrees (Table 3.2). The academic degrees are listed in Appendix 3.

Table 3.2. Academic degrees obtained in the projects in 2016.

Project acronym	Doctor	Master
MAPS		1
SAUNA	1	2
CASA		1
CATFIS	1	
INSTAB		1
KATVE		1
MONSOON		1*
USVA		1
WANDA		1
COMRADE		1
INFRAL		1**
Total	**2**	**11**

*) in collaboration with SADE project
**) in collaboration with INSTAB project

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).

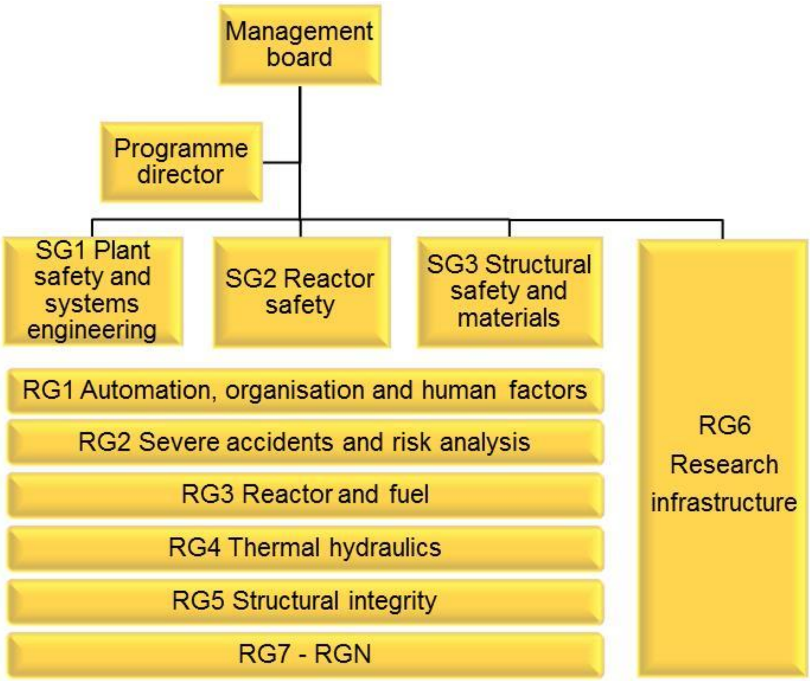


Figure 4.1. Structure of SAFIR2018 organisation. In 2016 each project belonged to one research area Steering Group (SG) and one Reference Group (RG). RG6 "Research infrastructure" has a special role of a steering and reference group [4].

During the administrative period (January 2016 – March 2017) the SAFIR2018 steering group held 4 meetings. Each of the steering groups SG-SG3 had 4 meetings and RG6 as a steering group 3 meetings. The reference groups RG1-RG6 had 3 meetings. The persons involved in the management board (MB) as well as the persons appointed by the MB to the steering and reference groups are listed in Appendix 5. Appendix 5 also shows the staff of the research projects and their main duties.

The SAFIR2018 management board can annually initiate small preliminary type studies with the order procedure. Decisions on the small sprojects are made after the funding decisions for the actual call for proposals. 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 2016 two projects were ordered and carried out: (1) Plant overall safety (LUT), and (2) Electrical systems (VTT, Aalto). The projects were formally realised as subcontracting in the administration project (ADMIRE). The final reports of the small projects can be found on SAFIR2018 extranet.

The programme director participated in the work of the Euratom Programme Committee (Fission configuration) as an expert member and two meetings of the national support group were also organised by SAFIR2018. The programme director also participated in the work of OECD NEA CSNI that had a meeting in December 2016.

The information on the research performed in SAFIR2018 was 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 interim seminar of SAFIR2018 was held on 23.-24.3.2017 at Innopoli. The seminar was for the first time 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 website (link). The SAFIR2018 Interim report [5] describing the work carried out in the programme during its two fist years can also be found on the website.

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Appendix 1

Publications in the projects in 2016

Crafting operational resilience in nuclear domain (CORE)

Scientific journal articles

Teperi, A.-M., Puro, V., & Ratilainen, H. (in press). Applying a new human factor tool in the nuclear energy industry. *Safety Science*, 95, 125-139.

Conference papers

Laarni, J., Karvonen, H., Pakarinen, S. & Torniainen, J. (2016). Multitasking and interruption management in control room operator work during simulated accidents. In *Proceedings of the HCI International 2016*. Springer.

Liinasuo, M., Koskinen, H., & Porthin, M. (2016). Principles, practises and developmental needs of emergency exercises in Finland" In *Proceedings of the Enlarged Halden Group Meeting (EHPG)*, May 8-13, Sandefjord, Norway.

Pakarinen, S., Korpela, J., Torniainen, J., Laarni, J & Karvonen, H. (2016). Control Room Operator Stress and Cardiac Activity in Simulated Incident and Accident Situations. In *Proceedings of Enlarged Halden Group Meeting (EHPG)*, May 8-13, Sandefjord, Norway.

Pakarinen, S. A., Korpela, J., & Torniainen, J. (2016). Quantifying acute stress with heart rate variability (HRV) and electrodermal activity (EDA) in real world conditions. *International Journal of Psychophysiology*, (108), 73-74.

Teperi, A.-M. (2016). Modifying human factor tool for work places - development processes and outputs. Poster presentation in *the 12th World Conference on Injury Prevention and Safety Promotion*, 18-21 September, 2016, Tampere, Finland.

Teperi, A.-M., Ratilainen, H., Puro, V., 2016. Need for new human factor models and tools in the safety-critical nuclear domain. Oral presentation in *the 12th World Conference on Injury Prevention and Safety Promotion*, 18–21 September, 2016 in Tampere, Finland.

Torniainen, J., Korpela, J. & Pakarinen, S. (2016). Control Room Operator Stress and Electrodermal Activity in Simulated Incident and Accident Situations. In *Proceedings of the Enlarged Halden Group Meeting (EHPG)*, May 8-13, Sandefjord, Norway.

Viitanen, K., Koskinen, H., Axelsson, C., Bisio, R., Linnasuo, M. & Skjerve, A. B. (2016). Learning from Successful Experiences: An Undeveloped Potential in the Nuclear Industry? In *Proceedings of the Enlarged Halden Group Meeting (EHPG)*, May 8-13, Sandefjord, Norway.

Wahlström, M. & Kuula., T. (2016). Organizational self-determination and new digital self-study applications as means for developing nuclear power plant operation training" In *Proceedings of HCI International 2016.*

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Viitanen, K., Bisio, R., Axelsson, C., Koskinen, H., Liinasuo, M., & Skjerve, A. B. (2016). Learning from Successes in Nuclear Power Plant Operation - Intermediate Report from the NKS-R LESUN (No. NKS-354). NKS. Retrieved from http://www.nks.org/en/nks_reports/view_document.htm?id=111010213330253

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Bergroth, J., Koskinen, H., & Laarni, J. (2016). Use of immersive 3D virtual reality environments in control room validations. Abstract submitted to *ANS NPIC&HMIT 2017.*

Laarni, J. (2017) Modelling collaborative troubleshooting in nuclear domain. A slide set.

Laarni, J. (2016). Modelling collaborative troubleshooting in nuclear domain. Abstract submitted to *ANS NPIC&HMIT 2017.*

Ratilainen, H., Teperi, A-M., Puro, V. 2017. How does the nuclear industry learn from operating experiences? WOS Congress in Praque 10/2017. Abstract submitted.

Skjerve, A.B., Viitanen, K., Axelsson, C., Bisio, R., Koskinen, H. & Liinasuo, M. (2016). Learning from Successes in Nuclear Operations – A Guideline to be presented at *ESREL2017* in 18.-22.6.2017.

Wahlström, M. et al. (2016). Supporting resilience in nuclear power plant operations - A summary of conceptual frames for human factors development and validation. Abstract submitted to *ANS NPIC&HMIT 2017.*

Wahlström M., Kuula, T., Seppänen, L. (2017). Resilient power plant operations through a self-evaluation method. Resilience engineering symposium 7th REA Symposium 26th-29th June 2017 Belgium Liege. Abstract submission accepted.

Extreme weather and nuclear power plants (EXWE)

Scientific journal publications

Gregow, H., Laaksonen, A. and Alper, M.E. 2017: Increasing large scale windstorm damage in Western, Central and Northern European forests, 1951-2010. Scientific Reports (in press).

Kämäräinen M, Hyvärinen O, Jylhä K, Vajda A, Neiglick S, Nuottokari J, Gregow H, 2017: A method to estimate freezing rain climatology from ERA-Interim reanalysis over Europe, Nat. Hazards Earth Syst. Sci., 17, 243-259, doi:10.5194/nhess-17-243-2017

Lehtonen I and Karpechko AYu, 2016. Observed and modeled tropospheric cold anomalies associated with sudden stratospheric warmings. J. Geophys. Res. Atmos., 121, 1591-1610. doi: 10.1002/2015JD023860

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Ukkonen P, A. Manzato, A. Mäkelä: Evaluation of thunderstorm predictors for Finland using reanalyses and neural networks. Journal of Applied Meteorology and Climatology (submitted)

Wintoft, P., Viljanen, A. and Wik, M., 2016. Extreme value analysis of the time derivative of the horizontal magnetic field and computed electric field. Ann. Geophys., 34, 485-491, doi:10.5194/angeo-34-485-2016.

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https://ams.confex.com/ams/97Annual/webprogram/Paper313860.html

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Kämäräinen M, Vajda A, Hyvärinen O, Lehtonen I, Jylhä K, 2016: Future projections of freezing rain climatology in Europe. EMS Annual Meeting Abstracts, Vol. 13, EMS2016-399-1, 2016, 16th EMS / 11th ECAC. http://meetingorganizer.copernicus.org/EMS2016/EMS2016-399-1.pdf

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Management principles and safety culture in complex projects (MAPS)

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Viitanen, K. and Reiman, T. (2016) How to build an "adaptive safety culture" in the nuclear industry? – A safety leadership approach, Extended conference abstract, accepted to the 7th Symposium of The Resilience Engineering Association (REA), 26-29 June 2017 (deliverable for 2016)

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Tuovinen, J. and Gotcheva, N. (2017) Delays in creating and handling design documents: A case study of a complex nuclear industry project, VTT Research Report, VTT-R-00892-17 (deliverable for 2016)

Viitanen, K., Gotcheva, N. and Rollenhagen, C. (2017) Safety Culture Assurance and Improvement Methods in Complex Projects – Intermediate Report from the NKS-R SC_AIM, NKS-381, January 2017. http://www.nks.org/en/nks_reports/view_document.htm?id=111010214063661 (deliverable for 2016)

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Probabilistic risk assessment method development and applications (PRAMEA)

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R. Pohja, S. Huotilainen, Report on an experimental campaign on cladding transient, VTT Research Report VTT-R-00799-17, 2017.

J. Kättö, SPACE – Validation tool for simulation software – User's guide for version 2.0.0, VTT Research Report, VTT-R-00204-17, 2017.

A. Arkoma, Validation status of the SCANAIR-GENFLO coupling, VTT Research Report, VTT-R-XXXX-XX, 2017.

Others

Travel reports

- NUMAT2016 conference
- Baltica X conference
- EHPG2016
- NINTH INTERNATIONAL CONFERENCE ON NUCLEAR AND RADIOCHEMISTRY – NRC9
- 155th HPG meeting and FSRM2016

Safety analyses for dynamical events (SADE)

Scientific journal publications

Ikonen, T., Syrjälahti, E., Valtavirta, V., Loukusa H., Leppänen, J. & Tulkki, V. Multiphysics simulation of fast transients with the FINIX fuel behaviour module. EPJ Nuclear Sci. Technol. 2, 37 (2016). http://dx.doi.org/10.1051/epjn/2016032

Conference papers

Sahlberg, V., Recalculating the steady state conditions of the V1000 zero power facility at Kurchatov Institute using Monte Carlo and nodal diffusion. Conference proceedings. Atomic energy Research (AER), pp. 485-497. 26th Symposium on VVER reactor physics and reactor safety, October 10-14, 2016, Helsinki, Finland. ISBN 978-963-7351-26-6, ISBN 978-963-7351-27-3.

Hovi, V., Taivassalo, V., Hämäläinen, A., Syrjälahti, E., Räty, H., Startup of a cold loop in a VVER-440, the 7th AER benchmark calculation with HEXTRAN-SMABRE coupled with porous CFD code PORFLO. Conference proceedings. Atomic energy Research (AER), pp. 369-389. 26th Symposium on VVER reactor physics and reactor safety, October 10-14, 2016, Helsinki, Finland. ISBN 978-963-7351-26-6, ISBN 978-963-7351-27-3.

Others

Hämäläinen, A., Simulation of the AER 7th benchmark with HEXTRAN-SMABRE-PORFLO, preliminary results. AER working group D meeting, May 30-31, 2016, Villingen, Switzerland. (contributed talk)

Sahlberg, V., Modelling of axial discontinuities in reactor cores with Serpent 2 - TRAB3D code sequence, Master’s Thesis, Aalto University. 59+6 p.
http://urn.fi/URN:NBN:fi:aalto-201608263123

Uncertainty and sensitivity analyses for reactor safety (USVA)

Scientific journal articles

Arkoma, Asko, Ikonen, Timo, Sensitivity analysis of local uncertainties in large break loss-of-coolant accident (LB-LOCA) thermo-mechanical simulations, Nuclear Engineering and Design Vol. 305, pp. 293-302.

Pusa, Maria, Isotalo, Aarno, Uncertainty analysis of assembly and core-level calculations with application to CASMO-4 and SIMULATE-3. Manuscript submitted to Annals of Nuclear Energy

Conference papers

Arkoma, Asko, Ikonen, Timo, Statistical and sensitivity analysis of failing rods in EPR LB-LOCA. In proceedings of: TopFuel 2016, Boise, Idaho, U.S.A., September 11-16, 2016, Paper no. 17570

Pusa, Maria, Uncertainty analysis of assembly and core-level calculations with application to CASMO-4 and SIMULATE-3, in proceedings of PHYSOR 2016.

Research reports

Alku, Torsti, Methods for the quantification of uncertainties related to Apros' physical models, VTT Research report, VTT-R-00571-17.

Arkoma, Asko, Uncertainty and sensitivity analysis methods in nuclear fuel modelling – a literature review, VTT Research Report, VTT-R-05086-16.

Arkoma, Asko, Applying support vector machines (SVMs) to predict fuel failures in LOCA, VTT Research Report, VTT-R-00964-17

Others

Taavitsainen, Aapo, 2016. CFENSS-SRS method for the uncertainty analysis of nuclear fuel and neutronics, Thesis for the degree of Master of Science in Technology, Aalto University, School of Science

Ikonen, Timo, Travel report on the LWR Uncertainty Analysis in Modelling (UAM)-10 benchmark workshop

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

Conference papers

Alexis Fedoroff, Juha Kuutti, Arja Saarenheimo, A physically motivated element deletion criteria for the concrete damage plasticity model, SMiRT24, August 20-25, 2017, Busan,Korea.

Arja Saarenheimo, Kim Calonius, Alexis Fedoroff, Markku Tuomala and Ari Vepsä, Experimental and Numerical Studies on Vibration Propagation, SMiRT24, August 20-25, 2017, Busan,Korea.

Arja Saarenheimo, Kim Calonius, Alexis Fedoroff, Markku Tuomala and Ari Vepsä, Numerical Sensitivity Studies on Vibration Propagation and Damping, SMiRT24, August 20-25, 2017, Busan,Korea.

Research reports

Vepsä, Ari, Impact testing of reinforced concrete slabs for combined shear punching and bending behaviour, VTT Research report, VTT-R-00267-17.

Fedoroff, Alexis, Continuum Damage Plasticity for Concrete Modeling, VTT Research report, VTT-R-00331-17.

Fire risk evaluation and Defence-in-Depth (FIRED)

Scientific journal articles

Topi Sikanen, Simo Hostikka: Predicting the Heat Release Rates of Liquid Pool Fires in Mechanically Ventilated Compartments. (accepted, spring 2017)

Conference articles and abstracts

Topi Sikanen: Simulation of liquid pool fires in mechanically ventilated compartments. Nordic fire safety days 16.-17-6-2016, Copenhagen.

Deepak Paudel: Propagation of model uncertainty in the presence of parameter uncertainty. FOMICS winter school. December 15-19, 2016. Lugano, Switzerland.

Research reports

Antti Paajanen, Jukka Vaari: Atomistic modelling of novel fire retardants. VTT-R-04781-16

Anna Matala: Ageing of flame retardant cables. VTT-R-05490-16.

Deepak Paudel and Simo Hostikka: Model uncertainty propagation in fire-barrier performance analyses.

Analysis of fatigue and other cumulative ageing to extend lifetime (FOUND)

Conference papers

Tommi Seppänen, Jouni Alhainen, Esko Arilahti and Jussi Solin, Direct Strain-Controlled Variable Strain Rate Low Cycle Fatigue Testing in Simulated PWR Water. ASME 2016 Pressure Vessels and Piping Conference, July 17–21, 2016, Vancouver, British Columbia, Canada. doi:10.1115/PVP2016-63294.

Aapo Ristaniemi, Linearization of supports with gaps in dynamic piping analyses. Baltica X - 2016 - Life Management and Maintenance for Power Plants.

Research reports

Otso Cronvall, Interim Report Draft 2 on: Susceptibility of BWR RPV and its internals to degradation mechanisms, Research Report, VTT-R-00368-17, Espoo, 2017.

Juha Kuutti, A J-integral calculation routine for Abaqus, VTT-R-05846-16, Espoo, 2017.

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Aapo Ristaniemi, Linearization of piping supports in dynamic response spectrum and time history analyses, VTT-R-01434-16, Espoo, 2016.

Qais Saifi, Computation procedure for FAC, general corrosion and cavitation-erosion, and their programming with in C++, Research Report, VTT-R-00152-17, Espoo, 2017.

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Antti Timperi, Study of Moment Combination Methods for Dynamic Analyses, Research report, VTT-R-00643-17, Espoo, 2017.

Antti Timperi, Spectrum Method for Modelling Crack Growth due to Thermal Mixing, VTT Research Report, VTT-R-00137-17, Espoo, 2017.

Tero Tyrväinen, Computation of consequences of piping component failures in PRA software, VTT-R-03666-16, Espoo, 2016.

Iikka Virkkunen, Measurement of residual stresses from BWR welds removed from service, Aalto University, Espoo, 2017-01-13.

Iikka Virkkunen, Comparison of test samples and residual stresses from BWR welds removed from service, Aalto University, Espoo, 2017-01-29.

Others

Juha Kuutti, FOUND2015: Review of results, VTT-R-01807-16, Espoo, 2016.

Tommi Seppänen. Research plan for studying the mechanisms and developing a new FEN model (PhD Thesis of Tommi Seppänen), Deliverable 3.2.2.

Long term operation aspects on structural integrity (LOST)

Scientific journal articles

Lindqvist, S. & Saifi. Q. 2017. Numerical simulations on crack growth in heat-affected zone of a VVER dissimilar metal weld. Engineering fracture mechanics. publication scheduled in November 2017

Sebastian Lindqvist, The effect of crack path on tearing resistance of a narrow-gap Alloy 52 dissimilar metal weld, Engineering fracture mechanics, Espoo, 2017. publication scheduled in November 2017

Conference papers

Sebastian Lindqvist, Tearing resistance of heterogeneous interface region of a dissimilar metal weld characterised with sub-sized single edge bend specimens, 21st European Conference on Fracture, ECF21, 20-24 June 2016, Catania, Italy.

Kim Wallin, Masato Yamamoto, Ulla Ehrnstén, Location of initiation sites in fracture toughness testing specimens – the effects of size and side grooves, Proceedings of the ASME 2016 Pressure Vessels and Piping Conference PVP2016, July 17-21, 2016, Vancouver, British Columbia, Canada.

Research reports

Heikki Keinänen, Stresses due thick overlay welding, Research Report, VTT-R-00374-17, Espoo, 2017.

Heikki Keinänen, Inlay welding as a nozzle repair method, literature survey of the residual stress computations, VTT-R-01073-16, Espoo, 2016.

Sebastian Lindqvist, Tommi Seppänen, BREDA: Fracture toughness measurements with miniature C(T) specimens in reference condition, Research Report, VTT-R-00140-17, Espoo, 2017.

Sebastian Lindqvist, Tearing resistance analysis of a VVER dissimilar metal weld mock-up, Research Report, VTT-R-03996-16, Espoo, 2016.

Sebastian Lindqvist, Dissimilar metal welds – the effect of crack path and specimen configuration on tearing resistance, Research Report, VTT-R-03998-16, Espoo, 2016.

Qais Saifi, Crack Growth Computation in Dissimilar Metal Weld Joints by Local Approach, Research Report, VTT-R-04464-16, Espoo, 2017.

Mitigation of cracking through advanced water chemistry (MOCCA)

Scientific journal publications

Essi Jäppinen, Tiina Ikäläinen, Sari Järvimäki, Timo Saario, Konsta Sipilä, Martin Bojinov, Effect of Octadecylamine on Carbon Steel Corrosion in Secondary Circuits of Pressurized Water Reactors, to be submitted for publication.

Conference papers

Essi Jäppinen, Tiina Ikäläinen, Timo Saario and Konsta Sipilä "Effect of Octadecylamine on Carbon Steel Corrosion Under PWR Secondary Side Conditions". Paper 62, 20th Nuclear Plant Chemistry International Conference, NPC 2016, October 2-7, 2106, Bournemouth, UK.

Essi Jäppinen, Timo Saario and Konsta Sipilä, "Improving passivation of carbon steel in steam cycles of power plants with a film forming amine". Baltica X - 2016 - Life Management and Maintenance for Power Plants International Conference. 7-9.6.2106, Finland and Sweden.

Research reports

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Martin Bojinov, Essi Jäppinen, Timo Saario and Konsta Sipilä, "Verification of a test method for lead assisted stress corrosion cracking of carbon steel", Research report VTT-R-05238-16

Essi Jäppinen, Konsta Sipilä and Timo Saario, "Effect of octadecylamine on carbon steel corrosion under PWR/VVER secondary side conditions – compilation of results from 2014 to 2016", Research report VTT-R-00312-17.

Others

Essi Jäppinen, Presentation: "Lead-induced stress corrosion cracking of carbon steel 22K", 12th Annual Meeting of ECG-COMON, June 13 and 14, 2016, SCK-CEN, Mol, Belgium

Thermal ageing and EAC research for plant life management (THELMA)

Scientific journal articles

Lucas, T., Forsström, A., Saukkonen, T., Ballinger, R., Hänninen, H. Effects of thermal aging on materials properties, stress corrosion cracking and fracture toughness of AISI 316L weld metal. Metallurgical Transactions A, Vol. 47A, August 2016, 3956-3970.

Mouginot, R., Sarikka, T., Heikkilä, M., Ivanchenko, M., Ehrnstén, U., Kim, Y.S., Kim, S.S., H. Hänninen, Thermal ageing and short-range ordering of Alloy 690 between 350 and 550 °C, Journal of Nuclear Materials. Vol 485, March 2017, Pp 56-66, available on-line December 2016.

Conference papers

Ivanchenko M. et al. Thermal aging induced phase transformations in nuclear grade type 316l stainless steel weld metal. Extended abstract in proceedings of "Material Issues in Design, Manufacturing and Operation of Nuclear Power Plants Equipment" MAINSTREAM-2016, 6-10.6.2016, St. Petersburg, Russia.

Ivanchenko M., Karlsen W., Karlsen T. Microstructural TEM examination of 2 dpa irradiated 304L stainless steel extracted from a tensile creep specimen from IFA-669. Proceedings of the Fuels & Materials Session of the 39th Enlarged Halden Programme Group Meeting, EHPGM 2016, 8 - 13.05.2016, Sandefjord, Norway.

Research reports

Hurley, C. Progress report on INCEFA+ (Horizon2020) for SAFIR2018 - THELMA. VTT-R- 5310-16, 40 p.

Toivonen, A. SAFIR2018 THELMA 2016 - ICGEAC Inconel 600 Round Robin progress at VTT in 2016. VTT-R-05170-16, 17p.

Others

Bjurman, M., Ivanchenko, M., Efsing, P. Ehrnstén, U. Hänninen, H. In Service Thermally Aged Cast Stainless Steel –Characterization by TEM and APT. Presentation at the EPRI International Light Water Reactors Material Reliability Conference 2016.

Ehrnstén, U. et al. MICRIN+ State of the art report on surface requirements and practises for NPP primary components. Nugenia+ report, February 2016.

Kilian R. et al. Draft NUGENIA Proposal for optimized surface conditions to mitigate in-service degradation (NUGENIA Position Paper), Nugenia+ report 27.09.2016.

NDE of NPP primary circuit components and concrete infrastructure (WANDA)

Scientific journal articles

Producing a POD curve with emulated signal response data, Koskinen, Tuomas; Virkkunen, Iikka; Papula, Suvi; Sarikka, Teemu; Haapalainen, Jonne

Conference papers

Artificial Flaw Detection with Ultrasound in Austenitic Stainless Steels, Koskinen, Tuomas, Leskelä, Esa; Vippola, Minnamari

PAUT Sizing Artificially Produced Fatigue Cracks in Austenitic Stainless Steel Weld, Koskinen, Ari; Leskelä, Esa

Ultrasonic Response on Artificially Produced Fatigue Cracks in AISI 321 Austenitic Stainless Steel Weld, Koskinen, Ari, Leskelä Esa

The effect of an austenitic weld to probability of detection of ultrasonic inspection, Koskinen, Tuomas; Haapalainen, Jonne

Measurements of the Extension of the Magnetite Pile on Steam Generator Tubing with Eddy Current Techniques, Jäppinen, Tarja; Ala-Kleme, Sanna

Selection Matrix for Non-Destructive Testing of NPP Concrete Structures, Al-Neshawy, Fahim; Ferreira, Miguel; Bohner, Edgar; Puttonen, Jari

NDE Research of Nuclear Power Plant Primary Circuit Components and Concrete Infrastructure in Finland, Jäppinen, Tarja; Ferreira, Miguel

Research reports

WANDA VTT-R-00645-17 Pre-design considerations for a large-scale NDE mock-up SAFIR 2018 – WANDA Project WP2 2016 Deliverable, Al-Neshawy, Fahim; Ferreira, Miguel; Bohner, Edgar

Others

Master's thesis: Artificial Flaw Detection with Ultrasound in Austenitic Stainless Steel, Koskinen, Tuomas

Travel report: (2016.02.15) IRSN-ODOBA_TripReport_FERREIRA

Travel report: (2016.10.10) ODOBA_TripReport_FERREIRA

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

Research reports

Granlund, Marcus. Development of condition monitoring technique for O-rings used in NPP applications, SP Research report, 6P01138.

Bondeson, Anna. Identification of available polymers and their data from Barsebäck, SP Research report, P601138-02

Antti Paajanen, Konsta Sipilä. Modelling tools for the combined effects of thermal and radiation ageing in polymeric materials, VTT Research report, VTT-R-00102-17.

Konsta Sipilä, Harri Joki. Synergistic effects of radiation and heat on EPDM and CSM rubber, VTT Research report, VTT-R-01179-17.

Konsta Sipilä, Harri Joki, Anna Jansson, Louise Wogelred. Oxidation of EPDM: oxidation depth measurements and effects on material properties, VTT Research report, VTT-R-01190-17.

Konsta Sipilä. Methods used in evaluating the severity of dose rate effect, VTT Research report, VTT-R-04773-16.

Others

Granlund, Marcus. Stort intresse när kärnkraftsindustrin amlades för att lära mer om åldring av polymerer. Nordisk Energi, number 5, 2016, pp. 62-64.

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Scientific journal articles

Patel, G., Tanskanen, V., Hujala, E., Hyvärinen, J., 2016. Direct contact condensation modeling in pressure suppression pool system. Nuclear Engineering and Design, available online. Linked also to the NURESA and INSTAB projects.

Conference papers

Ylönen, A., Varju, T., Hyvärinen, J., Estimation of Velocity Fields from the Axial Wire-Mesh Sensor Data. Specialist Workshop on Advanced Instrumentation and Measurement, Techniques for Nuclear Reactor Thermal Hydraulics, SWINTH-2016-#004, Livorno, Italy, June 15-17, 2016.

Research reports

Joonas Telkkä, INFRAL 1/2016: Research Based Requirements of MOTEL, Research report, Lappeenranta University of Technology, Nuclear Engineering, 2016.

Joonas Telkkä, Elina Hujala, Lauri Pyy, INFRAL 2/2016: Status report on the advances in thermal hydraulic measurements, Research report, Lappeenranta University of Technology, Nuclear Engineering, 2016.

JHR collaboration & Melodie follow-up (JHR)

Travel reports

Travel report of the 5th WG meeting

Travel report of the 6th WG meeting

Radiological laboratory commissioning (RADLAB)

Conference papers

Karlsen, Wade, The VTT Centre for Nuclear Safety, up-grading Finnish nuclear safety research, ATS Ydintekniikka, vol. 3, 2016, pg. 42.

Karlsen, Wade, The new VTT Centre for Nuclear Safety, Suomalaisen Ydintekniikan Päivät 2016 – SYP2016. http://www.ats-fns.fi/fi/syp2016/proceedings

Research reports

Moilanen, P., Paasila, M., Lappalainen, P., Water purification system for the Electric Discharge Machine (EDM), VTT Report, VTT-R-05845-15, 36 p.

Karlsen, W. and Paasila, M., Description of radioactive specimen storage database system, VTT Report VTT-R-00252-16

Moilanen, P., Lappalainen, P., Planman, T. and Lyytikäinen, T., Remote opening of an irradiated surveillance capsule by robot, VTT-R-03965-16, 20 p.

Tähtinen, S., VTT Hot Cell Design and Manufacturing by ITD – Project Quality Plan - Manufacturing, VTT Report, VTT-R-04150-16, 29 p.

Lydman, J., A three month visit at PSI, VTT Report VTT-R-03046-16

Others

2017 VTT Centre for Nuclear Safety End Users Seminar

Appendix 2

Participation in international projects and networks in 2016

Crafting operational resilience in nuclear domain (CORE):

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 (Nuclear Generation II & III Association)

Program Committee Memberships of International Conferences and Workshops:

HUMTOOL-project manager A-M Teperi participated as an invited participant and SMS and HF- expert in OECD/International Transportation Foundation, Safety Management Systems- Round Table at 23-24th March 2017. Participation added value to highlight the HF-work done at Finnish safety critical fields.

Extreme weather and nuclear power plants (EXWE):

FP7 project RAIN (Risk Analysis of Infrastructure Networks in response to extreme weather) (2014-2016)

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)

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.

AMAP (the Arctic Monitoring and Assessment Program) project AACA-C (Adaptation Actions for a Changing Arctic –part C)

The Nordic Council project ERMOND (Ecosystem resilience for mitigation of natural disasters) (2015-2017)

EUMETSAT Optical Lightning Imager (LI) Mission Advisory Group (MAG). FMI is an Invited Expert Institute.

Project Coordination in “Enhancing the MTG LI User Readiness in National Meteorological Services” (EUMETSAT funded project. Partner institutes IMGW (Poland), DWD (Germany), RMI (Belgium).

Finnish Nepalese Project (FNEP II), Kathmandu, Nepal. Funded by the Ministry of Foreign Affairs of Finland. EXWE-related thematic “Lightning location data usability in severe weather monitoring and early warning services in developing countries”.

PROMOSERV, Hanoi, Vietnam. Funded by the Ministry of Foreign Affairs of Finland. EXWE-related thematic “Lightning location data usability in severe weather monitoring and early warning services in developing countries”.

Finnish Pacific Project (FINPAC) 2013-2016, Pacific Independent Islands. Funded by the Ministry of Foreign Affairs of Finland. “Lightning location data usability in severe weather monitoring and early warning services in developing countries”.

Nepal World Bank Project 2014-2016, Nepal. Funded by World Bank. EXWE-related thematic “User requirements for a national lightning location network”.

Finnish Bhutanese Project, Bhutan. Funded by the Ministry of Foreign Affairs of Finland. EXWE-related thematic “Lightning location data usability in severe weather monitoring and early warning services in developing countries”.

Severe Storm Warning Service for Sri Lanka (SSWSS, Tekes-BEAM project). Supporting the early warning services for thunderstorms in Sri Lanka. 2016-2018.

ESA/Space Situational Awareness Programme. Contains products for geomagnetically induced currents in high-voltage power systems.

NASA Living With a Star (LWS) Institute GIC Working Group. Identifying, advancing, and addressing the open scientific and engineering questions pertaining to geomagnetically induced currents.

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:

- Arctic Climate Stability and Change
- High Impact Events and Climate Change

Pellikka, H.: Research visit to Institute of Oceanography and Fisheries, Split, Croatia, 1.3–1.4.2016. Collaboration with Croatian meteotsunami experts PhD Jadranka Šepić and Phd Ivica Vilibić on the topic "High-frequency sea-level oscillations on the Finnish coast and their connection to synoptic patterns"

May 2016

International co-operation in EXWE: In WP1, internationally developed and distributed observational (SYNOP), reanalysis (ERA-Interim) and climate model (CORDEX, CMIP5) data bases are heavily utilized. T1.1 benefits from previous research co-operation with US researchers. T1.2 benefits from studies of freezing rain by FMI in two on-going European research projects: EU/FP7 project RAIN (Risk Analysis of Infrastructure Networks in Response to Extreme Weather) and the Copernicus Climate Change Service (C3S) project CLIM4ENERGY (A service providing climate change indicators tailored for the energy sector).

The work in WP2 is strongly supported by collaboration with the Institute of Oceanography and Fisheries, Split, Croatia. Two one-month research visits from FMI to the Croatian institute have been made in 2015 and 2016.

Due to the short time allocated to WP3, there are no specific plans or needs for international collaboration. However, WP3 does utilise results of the EU/FP7 project EURISGIC (European Risk for Geomagnetically Induced Currents, coordinated by FMI in 2011-2014), which contained also estimations of extreme geomagnetic storms. A recent paper of extreme space weather events published together with the Swedish Institute of Space Physics can provide useful information to Finnish conditions.

The non-hydrostatic convection-permitting HARMONIE model, utilized both in WP1 and WP4, is being developed in the international HIRLAM and ALADIN consortia of European meteorological services. Findings in EXWE related to or based on HIRLAM will be relevant to these consortia.

Funding from the Swedish Radiation Safety Authority (SSM) will be received for tasks T1.2 and T3.1, and an agreement has been sent to SSM for signing.

Management principles and safety culture in complex projects (MAPS):

NKS project Safety Culture Assurance and Improvement Methods in Complex Projects (SC_AIM) in cooperation with KTH Royal Institute of Technology in Stockholm and the Swedish nuclear industry (Forsmark, OKG), as well as the Finnish nuclear industry (Fennovoima, Fortum)

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 OECD Nuclear Energy Agency (NEA), Division of Human Aspects of Nuclear Safety in terms of safety culture development

Co-operation with the Norwegium oil and gas industry, regulator and research (STATOIL, the Norwegian oil and gas regulator (PSA), University of Stavanger) in terms of management of complex projects and safety in both sectors

Co-operation with RMIT University, Australia and Brighton University, UK in terms of managing complex projects and project alliancing/collaborative project delivery arrangements

Co-operation with the Society for Risk Analysis, Nordic Chapter (president elected)

Co-operation with Korean nuclear society in terms of safety culture research (invitation for a special conference session on safety culture)

Probabilistic risk assessment method development and applications (PRAMEA):

NKS project L3PSA (Addressing off-site consequence criteria using Level 3 PSA) (Ilkka Karanta)

NKS project SPARC (Scenarios and Phenomena Affecting Risk of Containment Failure and Release Characteristics) (Ilkka Karanta, Tero Tyrväinen)

Cooperation with Swedish partners (Lloyd's Register Consulting, Risk Pilot) in multi-unit PRA (Kim Björkman, Tero Tyrväinen)

Cooperation with Nordic PSA Group project on HRA dependences in Task 2.3 HRA Dependencies and on use of HRA outside of PRA in Task 2.1 (Markus Porthin, Jan-Erik Holmberg)

Enlarged Halden Programme Group Meeting, 8th-13th May 2016, Fornebu Oslo, Norway

- Conference paper and presentation: Markus Porthin, Marja Liinasuo, Terhi Kling: Recent Development and Future Prospects in Human Reliability Analysis of Advanced Control Rooms in Nuclear Power Plants

FinPSA End User Group

OECD/NEA WGRISK (Working Group on Risk Assessment)

- Co-chairing of technical discussion on PSA experience in modelling digital I&C and presentation on Nordic experience from the MODIG project at the annual meeting of WGRISK (Markus Porthin)
- Participation in the activities on Status of Practice for Level 3 Probabilistic Safety Assessment and Status of Site Level PSA (Including Multi-unit PSA developments) (Ilkka Karanta, Markus Porthin)

ETSON (European Technical Safety Organizations Network), working group on Probabilistic Safety Assessment

Participation in group activities by email and meeting (Markus Porthin)

IAEA expert group for the development of the Safety Report on Human Reliability Assessment for Nuclear Installations

- Participation in group activities by email and meeting (Markus Porthin)

ISCH COST Action IS1304, Ahti Salo management committee member

Program Committee Memberships of International Conferences and Workshops:

- 3nd Nordic chapter of the Society for Risk Analysis Europe Risk Conference to be organised 1–2 November 2017, Espoo (Ahti Salo, Jan-Erik Holmberg).
- Member of the Programme Committee of the 21st International Federation of Operational Research Societies, 17-21 July 2017, Québec, Canada (Ahti Salo)

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

OECD/NEA Working Group on Risk Assessment (WGRISK), M. Porthin acted as the co-chair of the session in the WGRISK annual meeting March 2016 dedicated to PSA experience in modelling digital I&C.

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

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):

European Energy Research Alliance (EERA) Joint Programme Nuclear Materials (JPNM) steering committee

Participation in EERA JPNM pilot projects TASTE and PROMETEUS

European Sustainable Nuclear Industrial Initiative (ESNII) Task Force

IAEA (International Atomic Energy Agency) Information Exchange Meeting on SCWRs (Supercritical Water Cooled Reactor)

GIF (Generation IV InternationalForum) SCWR Materials & Chemistry (M&C) Project Management Board (PMB)

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)

Chemistry and transport of fission products (CATFIS):

OECD/NEA STEM-2 (Source Term Evaluation and Mitigation)

OECD/NEA BIP-3 (Behaviour of Iodine)

NKS project ATR (Impact of Aerosols on the Transport of Ruthenium in the primary circuit of nuclear power plant), co-operation with Chalmers University of Technology

NUGENIA TA2.4 Source term area

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

OECD/WGAMA (Working Group on Analysis and Management of Accidents)

OECD/HYMERES (Hydrogen Mitigation Experiments for Reactor Safety)

OECD/ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation)

FONESYS (The Forum & Network of System Thermal-Hydraulic Codes in Nuclear Reactor Thermal-Hydraulics)

USNRC/CAMP (Code Applications and Maintenance Program)

Couplings and instabilities in reactor systems (INSTAB):

NKS project COPSAR (Containment Pressure Suppression Systems Analysis for Boiling Water Reactors)

NORTHNET (The Nordic Thermal Hydraulics and Nuclear Safety Network) Roadmap 3

NURESAFE (Nuclear Reactor Safety Simulation Platform) (Euratom FP7 project)

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

OECD/NEA PKL Phase 3 project

OECD/NEA PKL Phase 4 project

Nuclear criticality and safety analyses preparedness at VTT (KATVE):

OECD/NEA NSC (Nuclear Science Committee)

OECD/NEA WPNCS (Working Party on Nuclear Criticality Safety)

OECD/NEA EGUNF (Expert Group on Used Nuclear Fuel)

OECD/NEA EGADSNF (Expert Group on Assay Data of Spent Nuclear Fuel)

AER WG E (Atomic Energy Research, working group E: radwaste, spent fuel and decommissioning)

EWGRD (European Working Group on Reactor Dosimetry)

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

Collaboration with International Serpent user community (650 users in 175 universities and research organizations in 37 countries worldwide.

Membership in the Executive Committee 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 2016 international conference.

Membership in the Technical Program Committee of the M&C 2017 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).

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

OECD/NEA HYMERES Panda HP1_6_2 CFD blind benchmark on the erosion of stratified helium layer.

Nordic Thermal Hydraulic Network (Northnet), Roadmap 1 (OpenFOAM CFD-solver for nuclear reactor safety assessment).

Nordic Thermal Hydraulic Network (Northnet), Roadmap 3 (Containment Pressure Suppression Systems Analysis for Boiling Water Reactors).

NKS project COPSAR (Containment Pressure Suppression Systems Analysis for Boiling Water Reactors).

Physics and chemistry of nuclear fuel (PANCHO):

OECD Halden Reactor Project

OECD/NEA Working Group on Fuel Safety

Halden Programme Group Fuel&Materials

Jules Horowitz Reactor Fuel Working Group

OECD/NEA – IRSN Cabri Water Loop Project, Technical Advisory Group (TAG)

OECD/NEA benchmark UAM (Uncertainty Analysis in Best-Estimate Modelling for Design, Operation and Safety Analysis of LWRs)

OECD/WGFS RIA benchmark Phase II

IAEA Coordinated Research Programme (CRP) Fuel modelling in accident conditions (FUMAC).

Safety analyses for dynamical events (SADE):

OECD/NEA WPRS Working Party on Scientific Issues of Reactor Systems

OECD/NEA EGUAM Expert Group on Uncertainty Analysis in Modelling

OECD/NEA Oskarshamn-2 (O2) BWR Stability Benchmark for Coupled Code Calculations and Uncertainty Analysis in Modelling

AER working group D on VVER safety analysis

Uncertainty and sensitivity analyses for reactor safety (USVA):

OECD/NEA UAM-LWR (Expert Group on Uncertainty Analysis in Modelling / Coupled Multi-physics and Multi-scale LWR analysis)

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

Participation in ERNCIP European Reference Network for Critical Infrastructure Protection, Thematic group Resistance of structures to explosion effects

Fire risk evaluation and Defence-in-Depth (FIRED):

OECD/NEA PRISME2

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 Beräkningsgupp (BG)

European Technical Safety Organization Network (ETSON)

Nugenia+: Project REDUCE (Justification of Risk Reduction through In-service Inspection)

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 committee meeting

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)

Visiting scientist Max Szolcek from Manchester University, working between 25.3 – 14.5.2016 at VTT, participating mainly in experiments on determination of surface charge of magnetite.

Thermal ageing and EAC research for plant life management (THELMA):

International Co-operative group on environmentally assisted cracking, ICG-EAC (U. Ehrnstén).

EU H2020 project INCEFA+ - Increasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment (C. Hurley).

Nugenia+ sub-project MICRIN+ - Mitigation of Crack Initiation + (U. Ehrnstén)

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):

Participate in the NDE Seminar on concrete pathologies, hosted by IRSN OECD/NEA
Participate in the ODOBA Technical and Agreement preparatory meetings, hosted by IRSN

Condition monitoring, thermal and radiation degradation of polymers inside NPP containments (COMRADE):

1st annual COMRADE workshop, 21st – 22nd September 2016, Borås, Sweden.

JHR collaboration & Melodie follow-up (JHR):

Jules Horowitz Reactor project:

- 5th meeting of the Working Groups, Tel Aviv, Israel
- 6th JHR Technical Seminar & Nugenia Forum, Marseille, France
- 6th meeting of the Working Groups, Mol, Belgium

Appendix 3

Academic degrees obtained in the projects in 2016

Crafting operational resilience in nuclear domain (CORE)

Doctor of Technology:

MSc Vuokko Puro has continued doctoral studies at the Tampere University of Technology (TUT), and as a part of that, was a writer in the reviewed scientific article in Safety Science, 95, 125-139 regarding results of HF tool application in nuclear industry. Also another HUMTOOL-researcher MSc Henriikka Ratilainen has continued her doctoral studies, with the theme of learning from OEs, which is handled in the abstract, submitted for WOS 10/2017 Praque.

Management principles and safety culture in complex projects (MAPS)

Master of Science

Starck, Matilda: Key dimensions of project governance and their relation to nuclear safety - an explorative study of nuclear industry projects, Master’s thesis, Department of Management and Organization, Hanken School of Economics, Helsinki, 2016 [available at Hanken intranet].

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

Doctor of Technology:

Jussi Lahtinen: Model checking large nuclear power plant safety system designs, Aalto University, Department of Computer Science, date of the defence 7.10.2016

Master of Science in Technology:

Joonas Linnosmaa: Structured safety case tools for nuclear facility automation, Tampere University of Technology, April 2016

Arttu Hirvonen: Implementing systems-theoretic process analysis (STPA) in safety-critical system design, University of Oulu, December 2016.

Comprehensive analysis of severe accidents (CASA)

Master of Science in Technology:

Magnus Strandberg: Coolability of porous core debris beds: Ex-Vessel Steam Explosion Analysis with MC3D. Aalto University, October 2016.

Chemistry and transport of fission products (CATFIS)

Doctor of Technology:

Kajan I., Transport and Containment Chemistry of Ruthenium under Severe Accident Conditions in a Nuclear Power Plant. 2016, (Chalmers University of Technology, Nuclear chemistry, Department of Chemistry and Chemical Engineering, SE-412 96 Göteborg, Sweden, ISBN 978-91-7597-464-4, Ny Serie Nr. 4145, ISSN nr: 346-718X, http://publications.lib.chalmers.se/publication/241683-transport-and-containment-chemistry-of-ruthenium-under-severe-accident-conditions-in-a-nuclear-power). This PhD thesis includes two scientific publications about ruthenium transport in a RCS study, which was performed in NKS-R collaboration between VTT and Chalmers (SAFIR2014 and SAFIR2018). The experiments were performed with VTT´s Ru transport facility.

Couplings and instabilities in reactor systems (INSTAB)

Master of Science in Technology:

Skripnikov Dmitry, Spray droplet size distribution measurement. Lappeenranta University of Technology, School of Energy Systems, Energy Technology, 2016, http://urn.fi/URN:NBN:fi-fe2016102625602, 69 p.

Nuclear criticality and safety analyses preparedness at VTT (KATVE)

Master of Science in Technology:

Toni Kaltiaisenaho: Implementing a photon physics model in Serpent 2, Aalto University, Department of Applied Physics, spring 2016 (Master’s thesis was prepared in KATVE 2015, graduation in spring 2016)

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

Master of Science in Technology:

Ville Sahlberg: Modelling of axial discontinuities in reactor cores with Serpent 2 - TRAB3D code sequence, Aalto University, Department of Applied Physics, 2016. (in collaboration with SADE).

Safety analyses for dynamical events (SADE)

Master of Science in Technology:

Ville Sahlberg, M.Sc. (Tech.), August 2016 (Aalto University)

Uncertainty and sensitivity analyses for reactor safety (USVA)

Master of Science in Technology:

Aapo Taavitsainen: CFENSS-SRS method for the uncertainty analysis of nuclear fuel and neutronics, Thesis for the degree of Master of Science in Technology, Aalto University, School of Science, 2016.

NDE of NPP primary circuit components and concrete infrastructure (WANDA)

Master of Science in Technology:

AALTO: Master Thesis by Vaglio Tessitore Giulia: “Designing thick walled concrete structures - Code of practice, requirements, and reinforcement” (still in progress during 2016)

Koskinen, Tuomas, VTT, march 2016

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

Master of Science in Technology:

Erik Bernmalm Törnström: Degradation of polymeric materials in nuclear applications – Study of the end of life criteria, Chalmers University of Technology, Chemistry and chemical engineering - Applied chemistry. Started on March 14th 2017 and finish estimated to be September 15th 2017.

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Master of Science in Technology:

Dmitry Skripnikov: Spray droplet size distribution measurement, Lappeenranta University of Technology, School of Energy Systems, Degree Program in Nuclear Engineering (Master's Thesis was prepared within the INFRAL and INSTAB projects)

Appendix 4

International travels in the projects in 2016

Crafting operational resilience in nuclear domain (CORE)

Koskinen, H., Liinasuo, M., Viitanen, K. Researcher workshop at Ringhals NPP, 30.9.2016.

Karvonen, H., HCI International 2016, July 17-22 2016, Toronto, Canada.

Liinasuo, M., EHPG (Enlarged Halden Programme Group Meeting) May 8-13 2016, Sandefjord, Norway.

Pakarinen, S., The 9th World Congress of International Brain Research Organization IBRO, July 7–11, 2016, Rio de Janeiro, Brazil.

Pakarinen, S., The 39th Enlarged Halden Research Project Meeting, May 9–12, 2016, Fornebu, Norway.

Torniainen, J., The 39th Enlarged Halden Research Project Meeting, May 9–12, 2016, Fornebu, Norway.

Wahlström, M., HCI International 2016, July 17-22 2016, Toronto, Canada.

Extreme weather and nuclear power plants (EXWE)

Gregow, H. 97th American Meteorological Society (AMS) Annual Meeting. Seattle, USA, 21.-27.1.2017.
https://ams.confex.com/ams/97Annual/webprogram/ataglance.html

Jylhä, K., Laurila T., Lehtonen, I., Mäkelä, A., Vajda, A. 16th EMS Annual Meeting & 11th European Conference on Applied Climatology (ECAC), Trieste, Italy, 12–16.9.2016, http://www.ems2016.eu/home.html

Laurila T. AGU Fall Meeting 12-16.12.2016, San Francisco, USA.

Leijala, U. Ocean Sciences Meeting, New Orleans, Louisiana, USA, 21.-26.2.2016. http://osm.agu.org/2016/

Leijala, U. European Geosciences Union General Assembly 2016, Wien, Austria, 17.-22.4.2016. http://www.egu2016.eu/

Leijala, U. AGU Fall Meeting 2016, San Francisco, California, USA, 12-16.12.2016. http://fallmeeting.agu.org/2016/

Pellikka, H. Research visit to Institute of Oceanography and Fisheries, Split, Croatia, 1.3.-1.4. 2016.

Management principles and safety culture in complex projects (MAPS)

Gotcheva, Nadezhda. Participation in the International Conference on Human and Organizational Aspects of Assuring Nuclear Safety – Exploring 30 Years of Safety Culture, 22-26 February 2016, IAEA HQ, Vienna, Austria.

Ylönen, Marja. Participation in the International Conference on Human and Organizational Aspects of Assuring Nuclear Safety – Exploring 30 Years of Safety Culture, 22-26 February 2016, IAEA HQ, Vienna, Austria.

Kujala, Jaakko. Participation in the European Academy of Management (EURAM), 1-4 June 2016, Paris, France.

Aaltonen, Kirsi. Participation in the European Academy of Management (EURAM), 1-4 June 2016, Paris, France.

Gotcheva, Nadezhda. Participation in the 12th World Conference on Injury Prevention and Safety Promotion, 18-21 September, Tampere, Finland

Ylönen, Marja. Norway, Stavanger, conducting interviews with Norwegium oil and gas industry representatives

Gotcheva, Nadezhda. Participation in AIM/NKS project workshop, Sweden, Stockholm,

Viitanen, Kaupo. Participation in AIM/NKS project workshop, Sweden, Stockholm,

Ylönen, Marja. Society for Risk Analysis Europe, Bath, United Kingdom: 20 - 22 June 2016.

Ylönen, Marja. ESREL 2016, Glasgow 25th – 29th September 2016.

Probabilistic risk assessment method development and applications (PRAMEA)

Markus Porthin. Enhanced Halden Program Group (EHPG) meeting, Sandefjord, 8-13 May, 2016.

Jan-Erik Holmberg. IAEA Consultancy Meeting on the development of the Safety Report on Human Reliability Assessment for Nuclear Installations, Vienna, 7-11 November, 2016.

Tyrväinen, T. 13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13) 2-7 October, 2016, Sheraton Grande Walkerhill, Seoul, Korea.

Ilkka Karanta. NPSAG/NKS level 3 PSA seminar, Solna, Sweden, 28 January, 2016.

A. Mancuso. Exchange research period under the supervision of professor Enrico Zio. Politecnico di Milano, Milan (Italy), 1 December 2015 – 1 May 2016.

A. Mancuso. EURO 2016, 28th European Conference on Operational Research. Poznan University of Technology, Poznan (Poland), 3-7 July 2016.

A. Mancuso. ESREL 2016, European Safety and Reliability. University of Strathclyde, Glasgow (Great Britain), 25-29 September 2016.

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

Papakonstantinou, N., Reliability and Maintainability Symposium (RAMS) 2016, January 25-28, 2016, Tucson, AZ, USA.

Porthin, M., 16th Annual Meeting of the CSNI Working Group on Risk Assessment (WGRISK), March 16-18, 2016, Paris, France.

Valkonen, J., 39th Enlarged Halden Programme Group Meeting (EHPG), May 8-13, 2016, Fornebu, Norway.

Buzhinsky, I., Vyatkin, V., 14th IEEE International Conference on Industrial Informatics (INDIN 2016), July 18-21, 2016, Futuroscope-Poitiers, France.

Vyatkin, V., Buzhinsky, I. Seminar on “Discrete state plant model generation from traces observed at the continuous plant model behavior” in ITMO University, September 1-3, 2016, St. Petersburg, Russia.

Pakonen, A., Vyatkin, V., 21st IEEE Conference on Emerging Technologies and Factory Automation (ETFA 2016), September 6-9, 2016, Berlin, Germany.

Varkoi, T., 10th International Conference on the Quality of Information and Communications Technology (QUATIC 2016), September 6-9, 2016, Lisbon, Portugal.

Hirvonen, A., 4th European STAMP Workshop 2016, September 13-14, 2016, Zürich, Switzerland.

Nevalainen, R., 23rd European Conference on Systems, Software and Services Process Improvement (EuroSPI 2016), September 14-16, 2016, Graz, Austria.

Lahtinen, J., 26th European Safety and Reliability Conference (ESREL 2016), September 25-29, 2016, Glasgow, UK.

Tyrväinen, T., 13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13), October 2-7, 2016, Seoul, Korea.

Pakonen, A., 9th International Workshop on Application of Field Programmable Gate Arrays in Nuclear Power Plant, October 3-6, 2016, Lyon, France.

Valkonen, J., EDF, France, October 10-11, 2016, Paris, France.

Valkonen, J., HPG meeting, and a visit to Tokyo Institute of Technology with IFE, October 16-22, 2016, Tokyo, Japan.

Valkonen, J., Halden in-kind working meeting, November 11-13, 2016, Halden, Norway.

Safety of new reactor technologies (GENXFIN)

Penttilä, S., European Sustainable Nuclear Industrial Initiative (ESNII) Task Force Meeting, 20.1.2016, Brussels, Belgium

Penttilä, S., IAEA (International Atomic Energy Agency) Information Exchange Meeting on SCWRs (Supercritical Water Cooled Reactor) & GIF (Generation IV International Forum) SCWR Materials & Chemistry (M&C) Project Management Board (PMB) meeting, 14-18.3.2016, Chengdu, China

Pohja, R., EERA JPNM pilot project TASTE: final technical meeting, 19.4.2016, Schiphol, Netherlands

Penttilä, S., European Energy Research Alliance (EERA) Joint Programme Nuclear Materials (JPNM) steering committee meeting, 1.6. 2016, Karlsruhe, Germany

Penttilä, S., European Sustainable Nuclear Industrial Initiative (ESNII) Task Force (TF) meeting, 15.6.2016, London, England

Penttilä, S., IAEA (International Atomic Energy Agency) Technical Meeting on Materials and Chemistry for Supercritical Water Cooled Reactors (SCWRs) & GIF (Generation IV International Forum) SCWR Materials & Chemistry (M&C) Project Management Board (PMB) meeting, 10-14.10.2016, CVR/Rez, Czech Republic

Penttilä, S., European Energy Research Alliance (EERA) Joint Programme Nuclear Materials (JPNM) steering committee meeting, 1.12.2016, Dresden, Germany

Tulkki, V., Energiforsk annual conference: Nuclear Technology and Policy Developments - a Global Perspective, 25.1.2017

Comprehensive analysis of severe accidents (CASA)

Takasuo, Eveliina. NUGENIA TA2 review meeting on in- and ex-vessel corium behaviour. 29 February - 4 March 2016. Yaiza, Spain.

Sevón, Tuomo. 3rd OECD BSAF-2 meeting. 5-8 July 2016. Tokyo, Japan.

Nieminen, Anna. THAI-3 PRG2 and MB2 meetings. 21-23 November 2016. Eschborn, Germany.

Sevón, Tuomo. CSARP/MCAP meeting. 12-16 September 2016. Bethesda, USA.

Sevón, Tuomo. 4th OECD BSAF-2 meeting. 9-13 January 2017. Paris, France.

Chemistry and transport of fission products (CATFIS)

Kärkelä, T., NKS-R project collaboration with Chalmers: visit in Göteborg, March 2016.

Kärkelä, T., Gouëllo, M., OECD/NEA STEM-2 “kick-off meeting” (Programme Review Group and Management Board), Paris, France, April 2016.

Gouëllo, M., Kärkelä, T., OECD/NEA BIP-3 “kick-off meeting” (Programme Review Group and Management Board), Paris, France, April 2016.

Gouëllo, M., International Congress on Advances in Nuclear Power Plants (ICAPP 2016) – San Francisco, USA, April 2016.

Kärkelä, T., 25th International Conference Nuclear Energy for New Europe, Portoroz, Slovenia, September 2016.

Kärkelä, T., 2016 ANS Winter Meeting and Nuclear Technology Expo, Las Vegas, USA, November 2016.

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

Hillberg S. USNRC/CAMP meeting, 1.11.-5.11.2016, Washington DC, USA.

Alku T. OECD/ATLAS PRG & MB meeting, 14.10-19.10.2016, Daejeon, South Korea.

Karppinen I. OECD/WGAMA meeting, 20.9.-23.9.2016, Paris, France.

Karppinen I. OECD/HYMERES PRG & MB, 27.6.-1.7.2016, Köln, Germany.

Kolehmainen J. OECD/HYMERES analytical workshop and discussions for continuation of the project, 27.6.-28.6.2016, Köln, Germany.

Kurki J. FONESYS meeting, 27.9.-29.9.2016, Eschborn, Germany.

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

Heikki Purhonen, Vesa Riikonen, Virpi Kouhia, The Programme Review Group and Management Board meetings of the OECD/NEA PKL Phase 3 Project, Lucca Italy, 12th April 2016.

Virpi Kouhia, Joint ATLAS – PKL Workshop, Lucca Italy, 13th – 15th April 2016.

Heikki Purhonen, Vesa Riikonen, Eetu Kotro, The kick-off meeting of the OECD/NEA PKL Phase 4 Project, Erlangen, Germany, 9th – 10th November 2016.

Nuclear criticality and safety analyses preparedness at VTT (KATVE)

Viitanen, T., PHYSOR 2016: Unifying Theory and Experiments in the 21st Century, May 1 - 5, Sun Valley, Idaho, USA

Leppänen, J., 6th International Serpent User Group Meeting, September 26 - 29, Milan, Italy

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

Leppänen, Viitanen, Valtavirta, Tuominen. Sun Valley, ID, USA, May 1-7, 2016 - International conference PHYSOR 2016,

Leppänen. New Orleans, LA, USA, June 12-16, 2016 - ANS Annual Meeting and Reactor Physics Division Executive Committee Meeting.

Leppänen, Viitanen, Valtavirta, Tuominen. Milan, Italy, Sep. 26-30, 2016 - 6th International Serpent User Group Meeting.

Leppänen. Las Vegas, NV, USA, Nov. 6-12, 2016. ANS Winter Meeting and Reactor Physics Division Executive Committee Meeting.

NOTE: Travel costs for PHYSOR 2016 and Serpent UGM were shared between multiple projects.

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

Pättikangas, T., Nordic Thermal Hydraulic Network (Northnet), Roadmap 1 Reference Group Meeting, 22 June, 2016, KTH, AlbaNova, Stockholm, Sweden.

Pättikangas, T., Nordic Thermal Hydraulic Network (Northnet), Roadmap 1 Reference Group Meeting, 14 December 2016, KTH, AlbaNova, Stockholm, Sweden.

Rämä, T., Nordic Thermal Hydraulic Network (Northnet), Roadmap 1 Reference Group Meeting, 22 June, 2016, KTH, AlbaNova, Stockholm, Sweden (in-kind contribution of Fortum).

Physics and chemistry of nuclear fuel (PANCHO)

A. Arkoma, 4th meeting of OECD/WGFS Task Group on RIA fuel rod code benchmark phase II, 20.-21.6.2016, Lucca, Italy

A. Arkoma, WGFS interim meeting, 21.6.2016, Lucca, Italy

A. Arkoma, FRAPCON/FRAPTRAN Users Group meeting, 11.9.2016, Boise, Idaho, U.S.A.

J. Kättö, 2nd RCM of FUMAC, 29.5.-3.6.2016, Vienna, Austria

H. Loukusa, NuMat2016: The Nuclear Materials Conference, 7.-10.11.2016, France.

V. Tulkki, 39th Enlarged Halden Programme Group Meeting, 8.-13.5.2016, Oslo, Norway

V. Tulkki, joint Fuel Safety Research Meeting 18.-19.10.2016, Mito, Japan and 155th Halden Programme Group meeting 20.-21.10.2016, Tokyo, Japan.

Safety analyses for dynamical events (SADE)

Hämäläinen, A., AER working group D, May 30-31, 2016, Villingen, Switzerland.

Uncertainty and sensitivity analyses for reactor safety (USVA)

Ikonen, T., LWR Uncertainty Analysis in Modelling (UAM)-10 benchmark workshop PSI, Villigen, Switzerland, 1.-3. June 2016.

Arkoma A., TopFuel 2016, Boise, Idaho, 13th September 2016.

Fire risk evaluation and Defence-in-Depth (FIRED)

Sikanen, Topi: Participation to OECD/NEA PRISME meeting 18-20.5.2016 in Aix-en-Provence, France.

Sikanen, Topi: Nordic fire safety days, 16.-17.6.2016, Aalborg University in Copenhagen.

Matala, Anna: Participation to OECD/NEA PRISME meeting 16.-18.11-2016 in Aix-en-Provence, France.

Simo Hostikka: FEMTC 2016, 16.-18.11.2016, Costa del Sol, Spain. FDS radiation model development planning meeting with Kevin McGrattan (NIST).

Deepak Paudel: FOMICS Winter school, uncertainty propagation. December 15-19, 2016 in Lugano, Switzerland.

Analysis of fatigue and other cumulative ageing to extend lifetime (FOUND)

Oinonen, A. NUGENIA TA8: ENIQ - Sub-Area Risk (SAR) meeting, 30.-31.3.2016, Senec, Slovakia.

Cronvall, O. Swedish-Finnish plant operators Beräkningssgrupp (BG) meetings in Forsmark, Sweden, 22.-25.5.2016

Cronvall, O. Baltica X - 2016 - Life Management and Maintenance for Power Plants. 7.-9.6.2016, Helsinki-Stockholm-Helsinki

Ristaniemi, A. Baltica X - 2016 - Life Management and Maintenance for Power Plants. 7.-9.6.2016, Helsinki-Stockholm-Helsinki

Seppänen, T. 2016 ASME PVP Conference July 17–21, 2016 Vancouver, Canada

Cronvall, O., NUGENIA TA8: ENIQ - Sub-Area Risk (SAR) meeting, 26.10.2016, Brussels, Belgium.

Cronvall, O. Swedish-Finnish plant operators Beräkningssgrupp (BG) meetings in several Swedish NPPs 27.11.-2.12.2016

Long term operation aspects of structural integrity (LOST)

Wallin, K. ASTM E08 fatigue and fracture committee meeting, 2-6th May, Grand Hyatt San Antonio; San Antonio, TX

Wallin, K. ASTM E08 fatigue and fracture committee meeting, 14-18th November, Renaissance Orlando at SeaWorld, 2016, Orlando, USA.

Lindqvist, S., ECF21, 21st European conference on fracture, 20-24th June, The Sheraton Catania Hotel & Conference Center, Catania, Italy.

Mitigation of cracking through advanced water chemistry (MOCCA)

Jäppinen, E., European Co-operative Group on Corrosion Monitoring (ECG-COMON) meeting, June 13-14, 2016, Mol, Belgium

Jäppinen, E., NPC 2016, October 2-7, 2016, Bournemouth, UK

Saario, T., International Cooperation Group on Environmentally Assisted Cracking of Water Reactor Materials (ICG-EAC) meeting, May 15-20, 2016, Qingdao China

Thermal ageing and EAC research for plant life management (THELMA)

Ivanchenko, Mykola. Enlarged Halden Programme Meeting, Sandefjord 9-13.5.2017

Ehrnsten, Ulla. International Co-operative Group on Environmentally Assisted Cracking, ICGEAC2016, Steering committee meetings and yearly meeting, Qingdao, China, 15-20.05-2016.

Ivanchenko, Mykola. MAINSTREAM conference, Zelenogorsk, Russia, 21-23.5.2016.

Mouginot, Roman. EPRI Alloy Primary Water Stress Corrosion Cracking Research Collaboration meeting 2016. USA, Tampa, 28.11. – 2.12.2016.

International travels in WANDA project in 2016

Miguel Ferreira, Edgar Bohner, Ari Koskinen, *NDE Seminar on concrete pathologies,* hosted by IRSN, 16th – 17th Feb. 2016, Cadarache/Aix-en-Provence, France -- Ari ei ja oliko muutkaan WANDA matkalla

Miguel Ferreira, IRNS Start-up meeting for ODOBA project, Aix-en-Provence, France

Ari Koskinen and Tuomas Koskinen 12th international conference on Non Destructive Evaluation in relation to structural integrity for nuclear and pressurized components, 3rd -7th October 2016, Croatia.

Tarja Jäppinen, Ari Koskinen and Fahim Al-Neshawy, World conference on Nondestructive testing, Munchen, 12th -17th June 2016

Tarja Jäppinen, Nugenia Forum2016, 5th -8th April 0216, Marseille, France

Tuomas Koskinen, Baltica X, 7th-9th June

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

Paajanen, A., Penttilä, S., Sipilä, K. COMRADE workshop, 21-22nd September, SP Sveriges Tekniska Forskningsinstitut, Borås, Sweden.

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Pyy Lauri & Telkkä Joonas. Research visit to University of Michigan, September 17–October 9, 2016, Ann Arbor, Michigan, U.S.

Kotro Eetu & Purhonen Heikki. Specialist Workshop on Advanced Instrumentation and Measurement Techniques for Nuclear Reactor Thermal Hydraulics, June 15–17, 2016, Livorno, Italy.

JHR collaboration & Melodie follow-up (JHR)

Huotilainen, S., Tulkki, V., *5th Meeting of the JHR Working Groups*, Tel Aviv, Israel, 9-10 February 2016

Huotilainen, S., Kotiluoto, P., *6th JHR Technical Seminar & Nugenia Forum*, Marseille, France, 5-7 April 2016

Huotilainen, S., Tulkki, V., *6th Meeting of the JHR Working Groups*, Mol, Belgium, 21-22 November 2016

Radiological laboratory commissioning (RADLAB)

Lydman, J., Extended researcher visit to Paul Scherrer Institute (PSI), 1st February - 31st May, 2016, Villigen, Switzerland.

Tähtinen, S., Kukkonen, A., Factory acceptance test of first hot-cell unit, 8-11th August, 2016, Dresden, Germany.

Karlsen, W., Tähtinen, S. Factory Acceptance test of second hot-cell unit, 25-28th October, 2016, Dresden, Germany

Lyytikainen, T., Tähtinen, S., Factory acceptance tests of Zwick-Roell mechanical testing devices, 28-30th November, 2016, Ettlingen and Ulm, Germany.

Tähtinen, S. Jokipii, M., Factory acceptance test of third hot-cell unit, 30th November-2nd December, 2016, Dresden, Germany.

Appendix 5

The management board, the steering groups, the reference groups and the scientific staff of the projects in 2016

SAFIR2018 Management Board – MB

(Status in June 2016)

Organisation	Member	Vice member
STUK	Marja-Leena Järvinen (Chair)	Risto Sairanen
STUK	Risto Sairanen (Vice chair)	Tomi Routamo
Aalto	Filip Tuomisto	Eila Järvenpää
Fennovoima	Hanna Virlander	Mia Ylä-Mella
Fortum	Kristiina Söderholm	Matti Kattainen
LUT	Juhani Hyvärinen	Heikki Purhonen
MEE	Jorma Aurela	Linda Kumpula
SSM	Nils Sandberg	N/A
Tekes	Arto Kotipelto	Reijo Munther
TVO	Liisa Heikinheimo	Risto Himanen
VTT	Eija Karita Puska	Petri Kinnunen
SAFIR2018 (Secretary)	Jari Hämäläinen	Vesa Suolanen

SAFIR2018 Steering Groups:

(Status in June 2016)

SG1 – Plant safety and systems engineering

Organisation	Member	Vice member
STUK	Tomi Routamo (Chair)	Eero Virtanen
Fennovoima	Pekka Viitanen	Juho Helander
Fortum	Karoliina Ekström	Leena Salo
TVO	Jari Pesonen (Vice chair)	Mikko Lemmetty
SAFIR2018 (Secretary)	Jari Hämäläinen	Vesa Suolanen

SG2 – Reactor safety

Organisation	Member	Vice member
STUK	Risto Sairanen (Chair)	Antti Daavittila
Fennovoima	Pekka Nurmilaukas	Juha Luukka
Fortum	Satu Sipola	Timo Toppila
TVO	Juha Poikolainen (Vice chair)	Matti Paajanen
SAFIR2018 (Secretary)	Jari Hämäläinen	Vesa Suolanen

SG3 – Structural safety and materials

Organisation	Member	Vice member
STUK	Martti Vilpas (Chair)	Pekka Välikangas
Fennovoima	Erkki Pulkkinen	Pasi Lindroth
Fortum	Ritva Korhonen	Ossi Hietanen
TVO	Timo Kukkola (Vice chair)	Paul Smeekes
SAFIR2018 (Secretary)	Jari Hämäläinen	Vesa Suolanen

SAFIR2018 – Reference Groups and Projects:

(Status in June 2016)

Reference Group	Projects	Comments
RG1 Automation, organisation and human factors	CORE (SG1), MAPS (SG1), SAUNA (SG1)	SG1 area
RG2 Severe accidents and risk analysis	EXWE (SG1), PRAMEA (SG1), GENXFIN (SG1), CASA (SG2), CATFIS (SG2), ERNEST (SG3), FIRED (SG3)	SG1, SG2 and SG3 areas
RG3 Reactor and fuel	KATVE (SG2), MONSOON (SG2), PANCHO (SG2), SADE (SG2)	SG2 area
RG4 Thermal hydraulics	COVA (SG2), INSTAB (SG2), INTEGRA (SG2), NURESA (SG2), USVA (SG2)	SG2 area
RG5 Structural integrity	COMRADE (SG3), FOUND (SG3), LOST (SG3), MOCCA (SG3), THELMA (SG3), WANDA (SG3)	SG3 area
RG6 Research infrastructure	INFRAL, JHR, RADLAB	RG6 area

RG1 – Automation, organisation and human factors

Organisation	Member	Vice member
Aalto	N/A	N/A
Fennovoima	Janne Peltonen (Vice chair), Sini Sipponen	Topi Tahvonen
Fortum	Juha Lamminen, Sami Matinaho, Leena Salo	Ville Nurmilaukas
LUT	Anne Jordan, Eetu Kotro	N/A
STUK	Mika Johansson, Pia Oedewald, Paula Savioja	Mika Kaijanen, Hanna Kuivalainen, Ann-Mari Sunabacka-Starck, Heimo Takala
TVO	Mauri Viitasalo (Chair), Petri Koistinen	Lauri Tuominen
VTT	Tommi Karhela, Heli Talja	N/A

RG2 – Severe accidents and risk analysis

Organisation	Member	Vice member
Aalto	Ahti Salo	N/A
Fennovoima	Juho Helander (Vice chair), Antti Paajanen, Janne Vahero	Leena Torpo
FMI	Heikki Tuomenvirta	Hilppa Gregow
Fortum	Calle Korhonen, Tapani Kukkola, Sami Siren	Tommi Purho
LUT	Jani Laine	N/A
STUK	Ellen Hakala, Ilkka Niemelä, Pekka Välikangas	Lauri Pöllänen
TVO	Antti Tarkiainen (Chair), Timo Kukkola, Lasse Tunturivuori	Maria Palomäki
VTT	Ilona Lindholm, Tony Rosqvist, Kim Wallin	N/A

RG3 – Reactor and fuel

Organisation	Member	Vice member
Aalto	Jarmo Ala-Heikkilä	N/A
Fennovoima	Jussi Kumpula, Kaisa Pellinen, Jukka Rintala	Libor Klecka
Fortum	Simo Saarinen, Laura Kekkonen	Jaakko Kuopanportti
LUT	Ville Rintala, Heikki Suikkanen	N/A
STUK	Antti Daavittila (Chair), Ulla Vuorio, Lena Hansson-Lyyra	N/A
TVO	Arttu Knuutila (Vice chair), Anssu Ranta-aho	Kari Ranta-Puska
VTT	Sami Penttilä, Joona Kurki	N/A

RG4 – Thermal hydraulics

Organisation	Member	Vice member
Aalto	Juhaveikko Ala-Juusela	N/A
Fennovoima	Juha Luukka, Jukka Lumela	Leena Torpo
Fortum	Timo Toppila (Vice chair), Aino Ahonen, Tapani Raunio	Tommi Rämä
LUT	Juhani Vihavainen, Lauri Pyy, Otso-Pekka Kauppinen	N/A
STUK	Eero Virtanen (Chair), Miikka Lehtinen	N/A
TVO	Janne Wahlman, Matti Paajanen	Timo Virtanen
VTT	Mikko Ilvonen, Anitta Hämäläinen, Jaakko Leppänen	N/A

RG5 – Structural integrity

Organisation	Member	Vice member
Aalto	Simo-Pekka Hannula, Iikka Virkkunen	N/A
Fennovoima	Mika Helin, Juha Rinta-Seppälä, Pasi Lindroth	Cem Ecevitoglu, Harri Lipiäinen
Fortum	Ossi Hietanen (Vice chair), Sanna Ala-Kleme, Alpo Neuvonen	Ritva Korhonen
LUT	Vesa Tanskanen	N/A
STUK	Mika Bäckström, Mirka Schildt	Jukka Härkölä
TVO	Erkki Muttilainen (Chair), Paul Smeekes, Vesa Hiltunen	Kimmo Tompuri
VTT	Kim Wallin, Aki Toivonen, Pertti Auerkari	N/A

RG6 – Research infrastructure

Organisation	Member	Vice member
STUK	Dina Solatie	Martti Vilpas
Fennovoima	Petri Sane	Jussi Leppänen
Fortum	Jyrki Kohopää (Chair)	Kristiina Söderholm
MEE	Jorma Aurela	Linda Kumpula
TVO	Liisa Heikinheimo (Vice chair)	Erkki Muttilainen
Aalto	Mikko Alava	Filip Tuomisto
LUT	Heikki Purhonen	Juhani Hyvärinen
VTT	Petri Kinnunen	Timo Vanttola

Project personnel

Crafting operational resilience in nuclear domain (CORE)

Research organisation: VTT, TTL
Project manager: Jari Laarni, VTT

Person	Org.	Tasks
Jari Laarni, PhD	VTT	Project manager, Human factors engineering, Work design, Cognitive modelling, Cognitive psychology
Hannu Karvonen, MA	VTT	Human factors, Functional situation modelling
Hanna Koskinen, MA	VTT	Human factors, Learning from successes, Emergency management
Timo Kuula, MA	VTT	Work-based learning, Work design
Marja Liinasuo, PhD	VTT	Human factors, Learning from successes, Emergency management
Markus Porthin, MScTech	VTT	Emergency management
Mikael Wahlström, PhD	VTT	Human factors, Work-based learning, Work design
Kaupo Viitanen	VTT	Safety culture, Learning from successes, Operating experience review
Satu Pakarinen, PhD	TTL	Deputy project manager, Psychophysiological methods, Stress management, Biofeedback
Kati Petterson, MScTech	TTL	Psychophysiological methods and analysis, Stress management
Vuokko Puro, MScTech	TTL	Human factors, Safety Management, Operational experience review
Henriikka Ratilainen, MScTech	TTL	Human factors, Safety Management, Operational experience review
Marika Schaupp, MA	TTL	Work-based learning, Work design

Laura Seppänen, PhD	TTL	Work-based learning, Work design
Anna-Maria Teperi, PhD	TTL	Human factors, Safety Management, Operational experience review
Maria Tiikkaja, PhD	TTL	Human Factors, Safety Management Systems
Kristian Lukander, MScTech	TTL	Psychophysiological methods and analysis, Stress management

Extreme weather and nuclear power plants (EXWE)

Research organisation: Finnish Meteorological Institute (FMI)
Project manager: Kirsti Jylhä, FMI

Person	Org.	Tasks
Kirsti Jylhä, Dr	FMI	Project management; WP1 coordination; freezing rain; sea-effect snowfall
Ari Venäläinen, Dr	FMI	Debyte project manager
Hanna Boman, MSc	FMI	Short-period sea level oscillations
Jan-Victor Björkqvist	FMI	Joint effect of high sea level and waves
Carl Fortelius, Dr	FMI	Fine scale numerical weather prediction
Hilppa Gregow, Dr	FMI	Freezing rain: contributing to writing
Marke Hongisto, Dr	FMI	Comparisons of model and obs data in WP4
Sebastian Heinonen	FMI	SILAM interface development
Otto Hyvärinen, Dr	FMI	Freezing rain: optimization methods
Milla Johansson, Dr	FMI	Joint effect of high sea level and waves
Ari Karppinen, Dr	FMI	Dispersion modelling; WP4 coordination
Anniina Korpinen	FMI	Fine scale numerical weather prediction
Ekaterina Kurzeneva, Dr	FMI	Fine scale numerical weather prediction
Matti Kämäräinen, MSc	FMI	Freezing rain
Terhi Laurila, MSc	FMI	Extreme convective weather; storm Mauri
Ilari Lehtonen, MSc	FMI	Synoptic analysis of meteotsunami cases
Ulpu Leijala, MSc	FMI	Joint effect of high sea level and waves
Anna Luomaranta, MSc	FMI	Sea-effect snowfall
Antti Mäkelä, Dr	FMI	Warm-season extreme convective weather; task 1.1 coordination
Taru Olsson, MSc	FMI	Sea-effect snowfall, HARMONIE runs
Havu Pellikka, MSc	FMI	Short-period sea level oscillations, WP2 coordination
Tuuli Perttula, MSc	FMI	Sea-effect snowfall, data assimilation
Jenni Rauhala, MSc	FMI	Warm-season extreme convective weather
Mikhail Sofiev, Dr	FMI	Dispersion modelling: development
Jani Särkkä, Dr	FMI	Joint effect of high sea level and waves
Andrea Vajda, Dr	FMI	Freezing rain: contributing to writing
Ari Viljanen, Dr	FMI	Space weather, WP3

Management principles and safety culture in complex projects (MAPS)

Research organisations: VTT, University of Oulu, Aalto University
Project manager: Nadezhda Gotcheva, VTT

Person	**Org.**	**Tasks and expertise**
Nadezhda Gotcheva, PhD	VTT	Project management, WP3 leader, WP5 leader, WP1 (safety culture expertise, literature review on project complexity, conceptual analysis, qualitative empirical data collection and analysis - interviews at the power companies and STUK, modelling of cultural complexity, internal integration, communication and dissemination of results)
Marja Ylönen, PhD	VTT	Deputy project management, WP2 leader (empirical data collection and analysis of the regulator's role in setting constraints and requirements for projects, benchmarking between the nuclear industry and the oil & gas industry- interviews and analysis)
Kaupo Viitanen	VTT	Task leader T3.3 (NKS SC_AIM project)
Sampsa Ruutu, PhD student	VTT	WP4 (system dynamics modelling)
Joona Tuovinen, PhD student	VTT	WP4 leader (system dynamics modelling)
Pertti Lahdenperä, Principle Scientist	VTT	Scientific support for WP1, construction industry network management, expert in alliance models/collaborative project delivery arrangements
Jaakko Kujala, Professor	University of Oulu	Project governance models of complex projects from safety point of view (WP1); system dynamics modelling support (WP4)
Kirsi Aaltonen, Assistant Professor	University of Oulu	Project governance models of complex projects from safety point of view (WP1); support for review on project complexity literature (WP1)
Karlos Artto, Professor	Aalto University	Scientific advisor (project business perspective); analysis of typical project governance models in megaprojects/complex projects
Matilda Starck, Master's student	Aalto University	Governance models in safety critical projects (WP1)

Probabilistic risk assessment method development and applications (PRAMEA)

Research organisation: VTT, Risk Pilot, Aalto
Project manager: Ilkka Karanta, VTT

Person	**Org.**	**Tasks**
Ilkka Karanta, LicTech	VTT	Project management, Level 2 PRA, Level 3 PRA method development and case studies
Kim Björkman, MScTech	VTT	Multi-unit PRA modelling, method support for level 2 PSA
Terhi Kling, MScTech	VTT	Requirements for HRA for advanced control rooms
Marja Liinasuo, DrPsych	VTT	Requirements for HRA for advanced control rooms, HRA outside of PRA
Teemu Mätäsniemi, MScTech	VTT	Method support for level 2 PSA
Markus Porthin, MScTech	VTT	Requirements for HRA for advanced control rooms, HRA outside the PSA
Tero Tyrväinen, MScTech	VTT	Multi-unit PRA modelling, Dynamic flowgraph methodology, IDPSA, Method support for levels 1 and 2 tight integration, Level 3 PSA method development and case studies
Jan-Erik Holmberg, DrTech	Risk Pilot	HRA outside of PRA, IAEA safety guide on HRA
Magnus Jacobsson, MScTech	Risk Pilot	Dependencies in HRA
Ahti Salo, DrTech, professor	Aalto	Project management, Reliability analysis of defence-in-depth in organizations
Alessandro Mancuso, MScTech	Aalto	Reliability analysis of defence-in-depth in organizations

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

Research organisation: VTT, Aalto University, Risk Pilot, FiSMA, IntoWorks
Project manager: Antti Pakonen, VTT

Person	Org.	Tasks
Antti Pakonen, MScTech	VTT	Project management, Closed-loop modelling in formal verification, Requirement editing and refinement for formal verification
Jarmo Alanen	VTT	Planning and management of qualification process and safety demonstration data
Kim Björkman, MScTech	VTT	Integration of model checking and PRA
Atte Helminen, MScTech	VTT	Modelling of digital I&C (MODIG)
Hanna Koskinen, MA	VTT	Structured safety demonstration of control room systems
Jussi Lahtinen, DrTech	VTT	Integration of model checking and PRA, Closed-loop modelling in formal verification
Jari Laarni, PhD	VTT	Structured safety demonstration of control room systems
Joonas Linnosmaa, MScTech	VTT	Guidelines for writing safety arguments, Planning and management of qualification process and safety demonstration data
Teemu Tommila, MScTech	VTT	Modelling for the early identification of weaknesses in DiD, Planning and management of qualification process and safety demonstration data, Guidelines for writing safety arguments
Nikolaos Papakonstantinou, DrTech	VTT	Modelling for the early identification of weaknesses in DiD
Markus Porthin, MScTech	VTT	Modelling of digital I&C (MODIG)
Tero Tyrväinen, MScTech	VTT	Modelling of digital I&C (MODIG)
Janne Valkonen, MScTech	VTT	Guidelines for writing safety arguments
Igor Buzhinskii, MSc	Aalto	Project management, Closed-loop modelling in formal verification, Requirement editing and refinement for formal verification
Valeriy Vyatkin, PhD	Aalto	Closed-loop modelling in formal verification, Requirement editing and refinement for formal verification
Jan-Erik Holmberg, DrTech	Risk Pilot	Project management, Modelling of digital I&C (MODIG)
Timo Varkoi, LicTech	FiSMA	Project management, Process assessment for systems and safety engineering

Risto Nevalainen, LicTech	FiSMA	Process assessment for systems and safety engineering
Eero Uusitalo, MScTech	IntoWorks	Project management, Application of STPA in digital I&C
Arttu Hirvonen, MScTech	IntoWorks	Application of STPA in digital I&C
Mika Koskela, MScTech	IntoWorks	Application of STPA in digital I&C

Safety of new reactor technologies (GENXFIN)

Research organisation: VTT
Project manager: Sami Penttilä, VTT

Person	Org.	Tasks
Sami Penttilä, M.Sc.	VTT	Project management, SMR material issues, International co-operation, Strategic views of future reactor concepts
Jarno Kolehmainen, M.Sc.	VTT	Safety features and licensing of SMRs
Aki Toivonen, D.Sc. (Tech.)	VTT	SMR material issues
Rami Pohja, M.Sc.	VTT	International co-operation
Ville Tulkki, D.Sc.(Tech.)	VTT	Strategic views of future reactor concepts
Tarja Jäppinen	VTT	SMR material issues
Pekka Nevasmaa	VTT	SMR material issues

Small Modular Reactor (SMR) training by Fortum: Sami Penttilä, Jarno Kolehmainen, Tarja Jäppinen

Comprehensive analysis of severe accidents (CASA)

Research organisation: VTT
Project manager: Anna Nieminen, VTT

Person	Org.	Tasks
Tuomo Sevón, , M.Sc. (Tech.)	VTT	OECD BSAF-2 project participation and U.S.NRC CSARP contact, Fukushima accident analyses with MELCOR, testing the new water ingression model in MELCOR.
Eveliina Takasuo, D.Sc. (Tech.)	VTT	Assessing debris bed coolability limit based on post-dryout temperature.
Veikko Taivassalo, D.Sc. (Tech.)	VTT	Comparing VTT’s MEWA results on debris bed post-dryout temperature behaviour to KTH’s DECOSIM results.
Magnus Strandberg, M.Sc. (Tech.)	VTT	Evaluating accident scenarios that could lead to hydrogen explosions in a Nordic BWR plant.
Anna Nieminen, M.Sc.	VTT	Project management, analysing in-

(Tech.)		containment dose rates and fission product behaviour with ASTEC and comparing the dose results with NRC method.
Jukka Rossi, M.Sc. (Tech.)	VTT	Defining in-containment dose rates with NRC method, extracting the ingestion dose coefficients from AGRID.
Mikko Ilvonen, Lic.Sc. (Tech.)	VTT	Implementing the ingestion dose pathways to VALMA.

Chemistry and transport of fission products (CATFIS)

Research organisation: VTT
Project manager: Teemu Kärkelä, VTT

Person	**Org.**	**Tasks**
Teemu Kärkelä, MScTech	VTT	Project management, Ruthenium, Iodine and HNO_3 experiments, Interpretation of results, Follow-up of OECD/NEA projects
Melany Gouello, PhD	VTT	Iodine experiments in primary circuit conditions, HNO_3 formation experiments, Follow-up of OECD/NEA projects
Jouni Hokkinen, MScTech	VTT	Iodine experiments in primary circuit conditions, Construction of experimental facilities
Karri Penttilä, MScTech	VTT	ChemPool calculations on pool pH
Tommi Kekki, MScTech	VTT	HNO_3 formation experiments
Petri Kotiluoto, PhD	VTT	HNO_3 formation experiments
Emmi Myllykylä, MSc	VTT	Chemical analysis - iodine experiments
Jaana Rantanen, Technician	VTT	Chemical analysis - iodine experiments
Tuula Kajolinna, Engineer	VTT	Analysis of gaseous compounds - iodine experiments

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

Research organisation: VTT
Project manager: Seppo Hillberg, VTT

Person	**Org.**	**Tasks**
Seppo Hillberg, M.Sc.(Tech)	VTT	Project manager, thermal-hydraulic analysis, nuclear power plant modelling, international cooperation/communication through research programmes (USNRC/CAMP)
Ismo Karppinen, M.Sc.(Tech)	VTT	thermal-hydraulic analysis, nuclear power plant modelling, international

		cooperation/communication through research programmes (OECD/WGAMA& HYMERES)
Jarno Kolehmainen, M.Sc.(Tech)	VTT	thermal-hydraulic analysis, nuclear power plant modelling, containment modelling, international cooperation/communication through research programmes (OECD/HYMERES)
Eric Dorval, D.Sc.(Tech.)	VTT	thermal-hydraulic analysis, nuclear power plant modelling
Ari Silde, M.Sc.(Tech)	VTT	thermal-hydraulic analysis, nuclear power plant modelling, containment modelling
Joona Kurki, Lic.(Tech)	VTT	international cooperation/communication through research programmes (FONESYS & OECD/ATLAS)
Torsti Alku, M.Sc.(Tech)	VTT	thermal-hydraulic analysis, nuclear power plant modelling, international cooperation/communication through research programmes (FONESYS)
Joona Leskinen, M.Sc.(Tech)	VTT	thermal-hydraulic analysis, nuclear power plant modelling
Sampsa Lauerma, student of technology	VTT	Research trainee in the area of nuclear power plant modelling

Couplings and instabilities in reactor systems (INSTAB)

Research organisation: LUT
Project manager: Markku Puustinen, LUT

Person	**Org.**	**Tasks**
Markku Puustinen, MScTech	LUT	Project manager, Experiment planning and analysis
Jani Laine, MScTech	LUT	Deputy project manager, Experiment analysis, Data conversion
Heikki Purhonen, DrTech	LUT	International tasks, Experiment planning
Vesa Riikonen, MScTech	LUT	Data acquisition, Experiments
Antti Räsänen, MScTech	LUT	Instrumentation, Data acquisition, Data conversion, Visualization, Control systems, Experiments
Vesa Tanskanen, DrTech	LUT	Computer simulations, Experiments
Giteshkumar Patel, MScTech	LUT	Computer simulations
Harri Partanen, Engineer	LUT	Designing of test facilities, Experiments
Hannu Pylkkö, Technician	LUT	Construction, operation and maintenance of test facilities, Experiments
Ilkka Saure, Technician	LUT	Construction, operation and maintenance of test facilities, Experiments

Lauri Pyy, MScTech	LUT	Assessment of measurement techniques, Experiments
Joonas Telkkä, MScTech	LUT	Assessment of measurement techniques, Experiments
Elina Hujala, MScTech	LUT	Pattern recognition, Experiment analysis
Eetu Kotro, MScTech	LUT	Construction, operation and maintenance of test facilities, Instrumentation, Data acquisition, Data conversion, Visualization, Control systems

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

Research organisation: LUT
Project manager: Vesa Riikonen, LUT

Person	**Org.**	**Tasks**
Vesa Riikonen, MScTech	LUT	Project manager, experiment planning, analysis and reporting, data acquisition
Markku Puustinen, MScTech	LUT	Deputy project manager
Heikki Purhonen, DTech	LUT	International tasks
Juhani Hyvärinen, DTech	LUT	International tasks
Virpi Kouhia, MScTech	LUT	APROS code modeling and calculations, system planning
Otso-Pekka Kauppinen, MScTech	LUT	TRACE code modeling and calculations
Jani Laine, MScTech	LUT	Internal report reviewer
Juhani Vihavainen, DTech	LUT	Internal report reviewer
Antti Räsänen, MScTech	LUT	Instrumentation, data acquisition, process control, experimental work
Harri Partanen, Engineer	LUT	Designing of test facilities
Hannu Pylkkö, Technician	LUT	Construction, operation and maintenance of test facilities, experimental work
Ilkka Saure, Technician	LUT	Construction, operation and maintenance of test facilities, experimental work
Eetu Kotro, MScTech	LUT	Construction, operation and maintenance of test facilities, experimental work

Nuclear criticality and safety analyses preparedness at VTT (KATVE)

Research organisation: VTT
Project manager: Tuomas Viitanen (-12/2016), Pauli Juutilainen (12/2016-), VTT

Person	Org.	Tasks
Tuomas Viitanen, D.Sc.(Tech.)	VTT	Project management, Neutron dosimetry, International co-operation, Burnup credit
Pauli Juutilainen, M.Sc.	VTT	Validation package, Burnup credit, Project management (12/2016-)
Ville Valtavirta, M.Sc.	VTT	Validation package, Gamma transport in Serpent
Riku Tuominen	VTT	Validation package, Gamma transport in Serpent
Silja Häkkinen, D.Sc.(Tech)	VTT	Burnup credit
Jaakko Leppänen, D.Sc.(Tech.)	VTT	Source terms, Gamma transport in Serpent, Neutron dosimetry
Antti Räty, MPhys	VTT	Activation analysis, QA Documentation & scientific publication
Petri Kotiluoto, PhD	VTT	International co-operation, Activation analysis
Risto Huhtanen, Lic.Tech.	VTT	Heat transfer in dry storage cask
Timo Pättikangas, D.Sc.(Tech.)	VTT	Heat transfer in dry storage cask
Juho Peltola, M.Sc.	VTT	Heat transfer in dry storage cask

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

Research organisation: VTT
Project manager: Jaakko Leppänen, VTT

Person	Org.	Tasks
Jaakko Leppänen, D.Sc.	VTT	Project manager, head developer of the Serpent code, development of methods for spatial homogenization and automated calculation sequence.
Ville Valtavirta, M.Sc.	VTT	Deputy project managed, Serpent developer, development of methods for assembly burnup calculations with fuel temperature feedback.
Maria Pusa, D.Sc.	VTT	Serpent developer
Tuomas Viitanen, D.Sc.	VTT	Serpent developer
Ville Sahlberg, M.Sc.	VTT	Serpent user, group constant generation for TRAB3D transient analysis code.
Antti Rintala, M.Sc.	VTT	Serpent user, group constant generation for the HEXBU-3D core simulator.

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

Research organisations: VTT, Aalto, LUT Energy, Fortum (in-kind contribution)
Project manager: Dr Timo Pättikangas, VTT

Person	**Org.**	**Task**
Timo Pättikangas, D.Sc.	VTT	Project manager, CFD modelling of PPOOLEX facility
Ville Hovi, M.Sc.	VTT	Coupled CFD-Apros simulations
Risto Huhtanen, Lic. Tech.	VTT	CFD modelling of PPOOLEX facility
Juho Peltola, M.Sc.	VTT	Two-phase CFD solver and model development for boiling heat transfer and validation
Joona Leskinen, M.Sc.	VTT	Coupled CFD-Apros simulations, Apros modelling
Ismo Karppinen, M.Sc.	VTT	Coupled CFD-Apros simulations, Apros modelling
Timo Siikonen, Prof.	Aalto	CFD model development and validation
Juhaveikko Ala-Juusela, M.Sc.	Aalto	CFD modelling of heat transfer in fuel rod bundles
Vesa Tanskanen, D.Sc.	LUT	CFD model development and validation for direct-contact condensation
Giteshkumar Patel, M.Sc.	LUT	CFD model development and validation for direct-contact condensation
Tommi Rämä, M.Sc.	Fortum	Validation of CFD solver for heat transfer in fuel rod bundles (in-kind contribution)
Timo Toppila, M.Sc.	Fortum	Validation of CFD solver for heat transfer in fuel rod bundles (in-kind contribution)

Physics and chemistry of nuclear fuel (PANCHO)

Research organisation: VTT
Project manager: Ville Tulkki, VTT; during 1.4.-31.7. Asko Arkoma, VTT

Person	**Org.**	**Tasks**
Asko Arkoma, MSc (Tech)	VTT	Analysis of reactivity initiated accidents and loss of coolant accidents, implementation and development of SCANAIR code.
Timo Ikonen, D.Sc. (Tech)	VTT	FINIX development
Joonas Kättö, MSc (Tech)	VTT	Improvement of validation database system, FINIX development
Henri Loukusa, MSc (Tech)	VTT	FINIX mechanical models development
Emmi Myllykylä, MSc	VTT	Leaching experiments on Thoria fuel.

Rami Pohja, MSc (Tech)	VTT	Cladding mechanical models, execution of experimental campaign
Ville Tulkki, D.Sc (Tech)	VTT	Development of cladding creep models, project manager.

Safety analyses for dynamical events (SADE)

Research organisation: VTT
Project manager: Ville Sahlberg, VTT since January 2017
Hanna Räty, VTT until end of 2016

Person		Task
Sahlberg Ville, MSc(Tech)*	VTT	Neutronics; project manager since January 2017
Hovi Ville, MSc (Tech)	VTT	3D thermal hydraulics
Hämäläinen Anitta, DSc (Tech)	VTT	HEXTRAN-SMABRE-PORFLO, International co-operation
Ilvonen Mikko, LicSc (Tech)	VTT	3D thermal hydraulics
Räty Hanna, MSc (Tech)	VTT	Project manager until end of 2016, neutronics, coupled calculations, International co-operation;
Syrjälahti Elina, MSc (Tech)	VTT	On parental leave since mid-February 2016 (project manager)
Taivassalo Veikko, PhLic (Phys)	VTT	3D thermal hydraulics

*) Graduated August 2016

Uncertainty and sensitivity analyses for reactor safety (USVA)

Research organisation: VTT
Project manager: Ville Valtavirta, VTT

Person	Org.	Tasks
Maria Pusa, D.Sc.(Tech.)	VTT	Project management, Nuclear data uncertainty propagation in full-core calculation sequences.
Asko Arkoma, M.Sc.	VTT	Project management, Analysis of rod failures in LB-LOCA.
Torsti Alku, M.Sc.	VTT	Methodology for determining input uncertainties
Aapo Taavitsainen, M.Sc. in 2016.	Aalto	Uncertainty and sensitivity analyses in multi-physics calculations.
Timo Ikonen, D.Sc.(Tech.)	VTT	Supporting role
Risto Vanhanen,	Aalto	Supporting role

D.Sc.(Tech.) in 2016		
Elina Syrjälahti, M.Sc.	VTT	Supporting role

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

Research organisation: VTT
Project manager: Ari Vepsä, VTT

Person	Org.	Tasks
Ari Vepsä, M.Sc.	VTT	Project management, Experimental testing
Alexis Fedoroff, D.Sc.(Tech)	VTT	Material modelling
Arja Saarenheimo, Lic.Tech.	VTT	Numerical simulation

Fire risk evaluation and Defence-in-Depth (FIRED)

Research organisation: VTT, AALTO
Project manager: Topi Sikanen, VTT (1-5/2016), Anna Matala, VTT (6-12/2016)

Person	Org.	Tasks
Topi Sikanen, MScTech	VTT	Project management, FDS development and validation.
Anna Matala, DrTech	VTT	Project management, Investigation of cable ageing
Antti Paajanen, MSc	VTT	Atomistic simulations of novel flame retardants
Jukka Vaari, DrTech	VTT	Atomistic simulations of novel flame retardants
Simo Hostikka, DrTech	Aalto	Project management, Barrier performance assessment with Fire-CFD,
Deepak Paudel, PhD student	Aalto	Barrier performance assessment with Fire-CFD, Uncertainty propagation between models.

Analysis of fatigue and other cumulative ageing to extend lifetime (FOUND)

Research organisation: VTT, Aalto
Project manager: Juha Kuutti, VTT

Person	Org.	Tasks
Juha Kuutti, M.Sc. Tech.	VTT	Project management; WP1 – Remaining lifetime and long term operation of components having defects; WP4 – Fatigue

		and crack growth caused by thermal loads
Otso Cronvall, Lic. Tech.	VTT	WP2 – Susceptibility of BWR RPV internals to degradation mechanisms; WP5 – Development of RI-ISI methodologies
Antti Timperi, M.Sc. Tech.	VTT	WP4 – Fatigue and crack growth caused by thermal loads; WP6 – Dynamic loading of NPP piping systems
Jouni Alhainen, M.Sc. Tech.	VTT	WP3 – Fatigue usage of primary circuit
Esko Arilahti, Res. Eng.	VTT	WP3 – Fatigue usage of primary circuit
Tommi Seppänen, M.Sc. Tech	VTT	WP3 – Fatigue usage of primary circuit
Jussi Solin, M.Sc. Tech	VTT	WP3 – Fatigue usage of primary circuit
Tero Tyrväinen, M.Sc. Tech	VTT	WP5 – Development of RI-ISI methodologies
Qais Saifi, M.Sc. Tech	VTT	WP5 – Development of RI-ISI methodologies
Ahti Oinonen, Dr. Tech.	VTT	WP5 – Development of RI-ISI methodologies; WP6 – Dynamic loading of NPP piping systems
Aapo Ristaniemi, M.Sc. Tech	VTT	WP6 – Dynamic loading of NPP piping systems
Iikka Virkkunen, Adj. Prof.	Aalto	WP7 – Residual stress relaxation in BWR NPPs

Long term operation aspects of structural integrity (LOST)

Research organisation: VTT
Project manager: Sebastian Lindqvist, VTT

Person	**Org.**	**Tasks**
Sebastian Lindqvist Project manager, Research scientist	VTT	Project management, Doctoral thesis on dissimilar metal welds (experimental), International co-operation,
Kim Wallin Deputy project manager, Research professor	VTT	International co-operation/project management
Päivi Karjalainen-Roikonen Senior Scientist	VTT	Doctoral thesis on fast fracture in upper shelf region
Qais Saifi Research scientist	VTT	Doctoral thesis on dissimilar metal welds (numerical)
Juha-Matti Autio Research scientist	VTT	Microstructural characterisation
Heikki Keinänen Senior scientist	VTT	Numerical analysis on residual stresses in dissimilar metal welds
Esa Varis Senior research engineer	VTT	Materials testing
Marke Mattila Senior research engineer	VTT	Materials characterisation

Jorma Hietikko Senior Research Technician	VTT	Materials testing
Tommi Seppänen, Research Scientist	VTT	Materials testing

Mitigation of cracking through advanced water chemistry (MOCCA)

Research organisation: VTT, University of Chemical Technology and Metallurgy (BG), Fortum Loviisa NPP
Project manager: Timo Saario, VTT

Person	Org.	Tasks
Timo Saario, D.Sc. (Tech)	VTT	Project management, data analysis, reports, scientific publication writing
Konsta Sipilä, MSc (Tech)	VTT	Experiments on effects of ODA, scientific publication writing
Essi Jäppinen, MSc (Tech)	VTT	Experiments on magnetite surface charge, scientific publication writing
Tiina Ikäläinen, BSc (Tech)	VTT	Water chemistry control, grab sample analysis
Seppo Peltonen, BSc (Tech)	VTT	Experimental design
Martin Bojinov, Prof., D.Sc. (Tech)	UCTM	Corrosion modelling, scientific publication writing
Sari Järvimäki, MSc (Tech)	Fortum	Plant data on N_2H_4 and ODA, scientific publication writing

Thermal ageing and EAC research for plant life management (THELMA)

Research organisation: VTT and Aalto
Project manager: Ulla Ehrnstén, VTT

Person	Org.	Tasks
Ulla Ehrnstén	VTT	Project management, mentoring, initiation testing
Mykola Ivanchenko Dr (Tech)	VTT	Characterisation of irradiated stainless steels, transmission electron microscopy
Juha-Matti Autio, M.Sc	VTT	Scanning electron microscopy, microstructural characterisation
Caitlin Hurley, Dr (Tech.)	VTT	fatigue initiation, electrochemical measurements
Aki Toivonen Dr (Tech)	VTT	Autoclave testing
Pekka Moilanen Dr (Tech)	VTT	Initiation testing of stainless steel material
Roman Mouginot doctoral student	Aalto	Thermal ageing of Alloy 690
Risto Ilola Dr (Tech)	Aalto	Project manager at Aalto
Hannu Hänninen professor	Aalto	Supervisor, mentoring

NDE of NPP primary circuit components and concrete infrastructure (WANDA)

Research organisation: VTT, Aalto
Project manager: Tarja Jäppinen, VTT

Person	Org.	Tasks
Tarja Jäppinen, Lic.Sc.(Tech.)	VTT	Project management, Eddy current modelling and testing
Miguel Ferreira, D.Sc.(Tech.)	VTT	Debuty project manager, Concrete infrastructure
Esa Leskelä, M.Sc.(Tech.)	VTT	Ultrasonic applications
Jonne Haapalainen, M.Sc.(Tech.)	VTT	Ultrasonic modelling, POD
Tuomas Koskinen, B.Sc.(Tech)	VTT	Ultrasonic applications, modelling, POD
Edgar Bohner, D.Sc.(Tech.)	VTT	Concrete infrastructure
Fahim Al-Neshawy D.Sc.(Tech.)	Aalto	Concrete infrastructure
Esko Sistonen D.Sc.(Tech.)	Aalto	Concrete infrastructure

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

Research organisation: VTT and SP
Project manager: Kosnta Sipilä, VTT

Person	Org.	Tasks
Konsta Sipilä, M.Sc.	VTT	Project management, polymer ageing mechanisms and effects inside NPP containments
Antti Paajanen, M.Sc.	VTT	Modelling tools for the synergistic effects of radiation and heat
Sami Penttilä, M.Sc.	VTT	Polymer ageing mechanisms and effects inside NPP containments
Harri Joki, M.Sc.	VTT	Polymer ageing during service failure, effects of radiation and heat on oxidation depth
Tiina Lavonen, M.Sc.	VTT	Development of condition monitoring methods for polymeric components including low dose rate radiation exposure
Marcus Granlund, M.Sc.	SP	Project management, development of condition monitoring methods for polymeric components including low dose rate radiation exposure
Anna Bondeson, M.Sc.	SP	Barsebäck pre-study
Anna Jansson, D.Sc.(Tech.)	SP	Effects of radiation and heat on oxidation depth

Louise Wogelred, M.Sc.	SP	Effects of radiation and heat on oxidation depth

Development of thermal-hydraulic infrastructure at LUT (INFRAL)

Research organization: LUT
Project manager: Joonas Telkkä, LUT

Person	**Org.**	**Tasks**
Joonas Telkkä, Project researcher	LUT	Project manager, WP1, WP2, WP3, WP4
Arto Ylönen, Post-doctoral researcher	LUT	Project manager until 3/2016, WP1, WP2, WP3, WP4
Markku Puustinen, Research scientist	LUT	WP1, WP2, WP3
Antti Räsänen, Research scientist	LUT	WP1, WP2, WP3
Heikki Purhonen, Research director	LUT	WP1, WP2, WP3
Virpi Kouhia, Project researcher	LUT	WP1, WP3
Jani Laine, Research scientist	LUT	WP1, WP2, WP3
Lauri Pyy, Project researcher	LUT	WP1, WP3
Harri Partanen, Design engineer	LUT	WP1, WP2, WP3
Eetu Kotro, Project researcher	LUT	WP1, WP2, WP3
Hannu Pylkkö, Technician	LUT	WP1, WP2, until 5/2016
Ilkka Saure, Technician	LUT	WP1, WP2
Elina Hujala, Doctoral student	LUT	WP1
Vesa Tanskanen, Post-doctoral researcher	LUT	WP1
Heikki Suikkanen, Post-doctoral researcher	LUT	WP1, WP3
Juhani Vihavainen, Research scientist	LUT	WP1, WP3
Otso-Pekka Kauppinen, Doctoral student	LUT	WP1, WP3
Ville Rintala, Doctoral student	LUT	WP1, WP3
Juhani Hyvärinen, Professor	LUT	WP1, WP2, WP3, WP4

JHR collaboration & Melodie follow-up (JHR)

Research organisation: VTT
Project manager: Santtu Huotilainen, VTT

Person	**Org.**	**Tasks**
Santtu Huotilainen, M.Sc. (Tech)	VTT	Project management, participation in the

		JHR collaboration as a member of the materials working group (MWG)
Petri Kinnunen, D.Sc. (Tech)	VTT	Participation in the JHR collaboration as the convenor of the technology working group (TWG)
Ville Tulkki, D.Sc. (Tech)	VTT	Participation in the JHR collaboration as a member of the fuel working group (FWG)

Radiological laboratory commissioning (RADLAB)

Research organisation: VTT
Project manager: Wade Karlsen, VTT

Person*	**Org.**	**Tasks**
Wade Karlsen, Ph. D.	VTT	Project Manager
Seppo Tähtinen, MSc	VTT	Lead, hot-cell technical realization
Arto Kukkonen, Tech	VTT	Hot-cell and equipment realization
Mika Jokipii, TechEng	VTT	Design engineering, EBW nuclearization
Ilkka Palosuo, MSc	VTT	Mechanical testing devices and shielding
Kimmo Rämö, Tech	VTT	Mechanicals
Marko Paasila, Tech	VTT	Database system development oversight
Matias Ahonen, DrTech	VTT	Investment portfolio oversight
Santtu Huotilainen, MSc	VTT	Waste handling systems design
Tommi Kekki, MScTech	VTT	Radiological safety
Ulla Vuorinen, MSc	VTT	Radiochemistry laboratory realization
Unto Tapper, DrTech	VTT	Microscopy infrastructure realization
Autio, Juha-Matti, MSc	VTT	Microscopy infrastructure realization
Jari Lydman, TechEng	VTT	Mechanical testing, remote handling
Mykola Ivanchenko, DrTech	VTT	Microscopy infrastructure realization
Aki Toivonen, PhD	VTT	Hot autoclave infra realization
Kirsti Helosuo, TechEng	VTT	Radiochemistry laboratory realization
Joonas Järvinen, MScTech	VTT	Radiochemistry laboratory realization
Marketta Mattila, Tech	VTT	Microscopy infrastructure realization

*Personnel making less than 1 person week of contribution are not listed