SAFIR2018 Annual Plan 2017

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Confidentiality: Public
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, concerning ensuring of expertise. 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 representatives of Radiation and Nuclear Safety Authority (STUK), Ministry of Employment and the Economy (MEE), 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).

In 2017 the planned volume of the SAFIR2018 programme is 6,7 M€ and 42 person years. Main funding organisations in 2017 are the Finnish State Nuclear Waste Management Fund (VYR) with 4,1 M€ and VTT with 1,4 M€. Research is carried out in 29 projects.

This report consists of a summary of the research plans and financial and administrative issues of the programme in 2017. The detailed research plans and budgets and personnel of the projects for 2017 are presented in Appendix 1. Appendix 2 contains the lists of members of the SAFIR2018 Management Board, Steering Groups and Reference Groups.
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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 a single licence holder.

The SAFIR2018 programme’s planning group, nominated by the Ministry of Employment and the Economy in March 2014, defined 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].

Several licensing and safety evaluation projects have been planned for the SAFIR2018 programme period or a time immediately following it: Olkiluoto 3’s operating license, Fennovoima’s construction license for Hanhikivi 1, the periodic safety review of Lovisa 1 and 2 plant units, and the renewal of the operating licenses of Olkiluoto 1 and 2 plant units. In addition, at the Lovisa and Olkiluoto plants, modernisation and improvement projects have been planned, including also significant automation renewals.

SAFIR2018 management board was nominated in September 2014. It consists of representatives of Radiation and Nuclear Safety Authority (STUK), Ministry of Employment and the Economy (MEE), 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 2017 was announced on the 26th of August 2016. After the closure of the call on the 21th of October 2016, SAFIR2018 management board, taking into account the evaluations made by the steering groups, prepared a proposal for the MEE regarding the projects to be funded in 2017. The funding decisions were made by the Finnish State Nuclear Waste Management Fund (VYR) in March 2017. In 2017 the programme consists of 29 research projects and a project for programme administration.
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 volume of the SAFIR2018 programme in 2017 is 6.7 M€ and 42 person years.

This annual plan summarises the research plans of the individual projects (Chapter 2), and provides financial statistics of the research programme (Chapter 3). Administrative issues are summarised in Chapter 4. The detailed research plans and budgets and personnel of the projects are given in Appendix 1. Appendix 2 contains lists of persons involved in the management board, research area steering groups and in the reference groups.

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. Research areas and research projects in 2017

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

1. Plant safety and systems engineering
2. Reactor safety

In addition, the development of research infrastructure is funded in the programme in order to ensure up-to-date research equipment and facilities. The research areas are presented with more detailed descriptions of their research needs for the programme period 2015-2018 in the SAFIR2018 Framework Plan [1].

In 2017, the research is carried out in 29 research projects. The planned total volume of the research projects is 6.4 M€. The volumes as well as the research execution organisations of the research projects in the above research areas are listed in Table 2.1.

Table 2.1 SAFIR2018 projects in 2017.

<table>
<thead>
<tr>
<th>Research area</th>
<th>Project Description</th>
<th>Acronym</th>
<th>Organisation(s)</th>
<th>Total funding (k€)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Plant safety and systems engineering</td>
<td>Crafting operational resilience in nuclear domain</td>
<td>CORE</td>
<td>VTT, FIOH</td>
<td>235.0</td>
</tr>
<tr>
<td></td>
<td>Extreme weather and nuclear power plants</td>
<td>EXWE</td>
<td>FMI</td>
<td>200.0</td>
</tr>
<tr>
<td></td>
<td>Management principles and safety culture in complex projects</td>
<td>MAPS</td>
<td>VTT, Aalto, University of Oulu</td>
<td>199.0</td>
</tr>
<tr>
<td></td>
<td>Probabilistic risk assessment method development and applications</td>
<td>PRAMEA</td>
<td>VTT, Aalto, Risk Pilot</td>
<td>328.0</td>
</tr>
<tr>
<td></td>
<td>Integrated safety assessment and justification of nuclear power plant automation</td>
<td>SAUNA</td>
<td>VTT, Aalto, FISMA, Risk Pilot, IntoWorks</td>
<td>349.0</td>
</tr>
<tr>
<td></td>
<td>Safety of new reactor technologies</td>
<td>GENXFIN</td>
<td>VTT</td>
<td>94.0</td>
</tr>
<tr>
<td></td>
<td>Electric Systems and Safety in Finnish NPP</td>
<td>ESSI</td>
<td>VTT, Aalto</td>
<td>130.0</td>
</tr>
<tr>
<td>2. Reactor safety</td>
<td>Comprehensive analysis of severe accidents</td>
<td>CASA</td>
<td>VTT</td>
<td>214.0</td>
</tr>
<tr>
<td></td>
<td>Chemistry and transport of fission products</td>
<td>CATFIS</td>
<td>VTT</td>
<td>143.0</td>
</tr>
<tr>
<td></td>
<td>Comprehensive and systematic validation of independent safety analysis tools</td>
<td>COVA</td>
<td>VTT</td>
<td>249.0</td>
</tr>
<tr>
<td></td>
<td>Couplings and instabilities in reactor systems</td>
<td>INSTAB</td>
<td>LUT</td>
<td>162.0</td>
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<tr>
<td></td>
<td>Integral and separate effects tests on thermal-hydraulic problems in reactors</td>
<td>INTEGRA</td>
<td>LUT</td>
<td>357.0</td>
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<tr>
<td></td>
<td>Nuclear criticality and safety analyses preparedness at VTT</td>
<td>KATVE</td>
<td>VTT</td>
<td>200.0</td>
</tr>
<tr>
<td></td>
<td>Development of a Monte Carlo based calculation sequence for reactor core safety analyses</td>
<td>MONSOON</td>
<td>VTT</td>
<td>146.0</td>
</tr>
</tbody>
</table>
### Development and validation of CFD methods for nuclear reactor safety assessment
- **NURESA** VTT, LUT 234.0

### Physics and chemistry of nuclear fuel
- **PANCHO** VTT 259.0

### Safety analyses for dynamical events
- **SADE** VTT 93.0

### Uncertainty and sensitivity analyses for reactor safety
- **USVA** VTT 115.0

### 3. Structural safety and materials

<table>
<thead>
<tr>
<th>Project Description</th>
<th>Team</th>
<th>Institution(s)</th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experimental and numerical methods for external event assessment improving safety</td>
<td><strong>ERNEST</strong></td>
<td>VTT</td>
<td>115.0</td>
</tr>
<tr>
<td>Fire risk evaluation and Defence-in-Depth</td>
<td><strong>FIRE</strong></td>
<td>VTT, Aalto</td>
<td>201.0</td>
</tr>
<tr>
<td>Analysis of fatigue and other cumulative ageing to extend lifetime</td>
<td><strong>FOUND</strong></td>
<td>VTT, Aalto</td>
<td>325.0</td>
</tr>
<tr>
<td>Long term operation aspects of structural integrity</td>
<td><strong>LOST</strong></td>
<td>VTT</td>
<td>307.0</td>
</tr>
<tr>
<td>Mitigation of cracking through advanced water chemistry</td>
<td><strong>MOCCA</strong></td>
<td>VTT</td>
<td>136.0</td>
</tr>
<tr>
<td>Thermal ageing and EAC research for plant life management</td>
<td><strong>THELMA</strong></td>
<td>VTT, Aalto</td>
<td>234.0</td>
</tr>
<tr>
<td>Non-destructive examination of NPP primary circuit components and concrete infrastructure</td>
<td><strong>WANDA</strong></td>
<td>VTT, Aalto</td>
<td>160.1</td>
</tr>
<tr>
<td>Condition monitoring, thermal and radiation degradation of polymers inside NPP containments</td>
<td><strong>COMRADE</strong></td>
<td>VTT, SP</td>
<td>188.0</td>
</tr>
</tbody>
</table>

### 4. Research infrastructure

<table>
<thead>
<tr>
<th>Project Description</th>
<th>Team</th>
<th>Institution(s)</th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>Development of thermal-hydraulic infrastructure at LUT</td>
<td><strong>INFRAL</strong></td>
<td>LUT</td>
<td>284.0</td>
</tr>
<tr>
<td>JHR collaboration &amp; Melodie follow-up</td>
<td><strong>JHR</strong></td>
<td>VTT</td>
<td>29.0</td>
</tr>
<tr>
<td>Radiological laboratory commissioning 2017</td>
<td><strong>RADLAB</strong></td>
<td>VTT</td>
<td>703.0</td>
</tr>
</tbody>
</table>

### 0. Programme administration

<table>
<thead>
<tr>
<th>Project Description</th>
<th>Team</th>
<th>Institution(s)</th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>SAFIR2018 administration</td>
<td><strong>ADMIRE</strong></td>
<td>VTT</td>
<td>355.0</td>
</tr>
</tbody>
</table>
2.1 Plant safety and systems engineering

In 2017 the research area “Plant safety and systems engineering” includes seven projects:

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

2.1.1 CORE – Crafting operational resilience in nuclear domain

In general, the aim of the project is to improve safe operation of nuclear power plants by developing guidance, good operational practices and other practical solutions that promote resilience in nuclear operations. In order to reach this objective, new guidelines, models and simulations, and training interventions that will be tested and examined in simulated test environments, will be developed.

The main objectives of the project are the following:

- understand how to capture successful performance adaptations from operating events and how to analyse and communicate the lessons learned from successes
- develop new concepts to describe operator work and develop means for promoting reflectivity and work based learning
- investigate the role of multitasking and effects of interruptions in operative work
- analyse the reasoning and problem solving processes in difficult troubleshooting situations
- promote self-reflection in training by self-assessment
- investigate the effects of acute stress on operator performance in simulated accident situations
- understand how emergency exercises are planned and conducted and investigate the resilience markers and developmental needs, especially the ones related to communication and coordination, in emergency exercises
- study appropriateness and use of the HF tool as an investigation method in operative event analysis
- evaluate the implementation process of the HF tool, its effects on safety thinking, as well as supporting and hindering factors of the process.

The project is divided into six parallel work packages, and each of them is further divided into several tasks (see Appendix 1). The work packages are associated with different levels of defence-in-depth. The work packages are the following:

1) Learning from successes in nuclear power plant operation to enhance organizational resilience
2) Developing work-based learning in the NPP domain
3) Supporting operational resilience in complex and dynamic environments
4) Supporting operator performance in extreme stress
5) Supporting resilience in emergency management
6) Applying a HF tool to learn to analyse human contribution to nuclear safety.
2.1.2 EXWE – Extreme weather and nuclear power plants

The objective of the research is to enhance scientific understanding of the environmental conditions of the NPP locations and predicting how they can change. By clearly explaining the methods and dataset fusions we enable replicability of the work and increase reliability in the results. EXWE primarily focuses on extreme weather and sea-level events that affect the design principles of the power plants and might pose external threats to the plants. A specific focus is given to extreme warm- and cold-season convective weather, including tornadoes and downbursts; wind-related multiple events, freezing rain; and extreme-sea-level events such as meteotsunamis. In addition, the project aims to improve the estimates of solar-storm effects on the critical Finnish infrastructure, in particular nuclear power plants. The work focusing on atmospheric dispersion modelling aims to provide a modern platform for assessing consequences of accidental releases at multiple transport and time scales. A spectrum of different sources of information will be utilized as the research material.

The expected results include:

- Frequency and trends of extreme convective weather conditions near the coasts of Finland.
- Frequency and trends of freezing rain at the NPP sites near the Finnish coast.
- Frequency and trends of combined weather events related strong winds near the Finnish coast.
- Exceedance frequency distribution of meteotsunamis at the Finnish NPPs.
- Combined probability distribution of high sea level and high waves on the Finnish coast.
- Follow-up of new knowledge regarding sea-level rise and, when necessary, updates to the previously calculated sea-level scenarios on the Finnish coast up to 2100.
- An integrated dispersion and dose-assessment toolset (SILAM Dose Assessment Post-Processor DAPP)
2.1.3 MAPS – Management principles and safety culture in complex projects

The ultimate goal of MAPS project is to enhance nuclear safety by supporting high quality execution of complex nuclear industry projects, including modernisations and new builds. Three objectives are formulated according to this ultimate goal, as follows (Fig. 2.1.3.1):

- Identify the generic safety principles of managing complex projects in the nuclear industry.
- Clarify the cultural phenomena in major projects and the influence of time, scale, governance models, and the diversity of the involved actors on safety culture, and thus on safety.
- Develop practical tools and guidance to facilitate the management and safety culture of ongoing and planned major projects focusing on, for example facilitating communication, managing change, organising decision making and problem solving in unexpected situations, encouraging openness, and distributing knowledge and lessons learned.

The project brings together expertise in nuclear safety culture, governance of complex projects, construction industry network management, societal research on safety regimes and system dynamics modelling. The overall expected result is advanced knowledge by developing a set of theoretical frameworks, guidance and practical tools for defining and assessing project management practices and safety culture enhancement and assurance in nuclear industry modernisations and new build projects.
2.1.4 PRAMEA – Probabilistic risk assessment method development and applications

Probabilistic risk assessment (PRA) is the field of quantifying risks in terms of probabilities, evaluating the contribution of different subsystems, processes etc. to total system risk, and assessing the uncertainty related to the analyses. Currently, as modernisation takes place (also) in the nuclear domain, the functional principles and practices both in the process and its control change. Modernisation raises the question of the existence of the needs to renew the practices and perhaps also the principles of PRA used in the nuclear power plant (NPP) in question. Furthermore, new builds in the nuclear domain call for tailored PRA.

The PRAMEA project will cover the important and topical issues in probabilistic risk/safety assessment for nuclear power plants. The main objectives of the project are as follows:

- to enable a credible human reliability analysis for digitalized control rooms in an acceptable, reliable and unified manner, to obtain new information on other topical issues and to train new HRA experts.
- develop methods that enable the incorporation of the probability of an emergency operation success, and timing information, to the main PRA models of nuclear power plants, and the planning of more cost-effective emergency operations with optimal use of scarce resources.
- Improve and develop methods to support a more extensive and practical overall safety assessment, and improve algorithms for computational PRA.
- develop a deeper understanding of some important serious accident phenomena (e.g. steam and hydrogen explosions) and how their analysis connects to the general framework of level 2 PRA.
- develop methods of uncertainty handling on level 3 PRA, improved ways of modelling important level 3 issues (such as evacuation), and software to aid level 3 PRA analyses.
- develop methods for site risk analysis, including development of risk criteria for site risk analysis and methods to analyse multi-unit accident scenarios.
- develop methods and models for the analysis of schedule and end-product quality risks in emergency operations so that estimates can be calculated for PRA-relevant quantities such as success probability of operation and the timing of operation completion.
- analyse the risks associated with an organization performing activities to attain a goal; for example, the risk that an error in a plan will not be detected in reviews and other quality assurance activities, and will affect the implementation of the plan.
- study applications and computational issues of dynamic flow graph methodology.
- train new PRA experts.
- develop cooperation with Finnish experts.
- foster international collaboration.
2.1.5 SAUNA – Integrated safety assessment and justification of nuclear power plant automation

The aim of the SAUNA project is to develop integrated and multidisciplinary ways to build confidence in the safety of nuclear power plants and their systems. Within the general framework of Systems Engineering (SE), this goal is pursued by developing comprehensive and transparent safety demonstration practices. The results, intended to be used by the regulator, the utilities and by the system suppliers, cover co-use of diverse assessment methods, structured representation of the safety justification arguments and common working processes for system development and licensing. While scoping the project primarily on plant operations, i.e., on instrumentation and control systems and their human users, the SAUNA team considers a power plant and organisations involved in its development, licensing and maintenance as a sociotechnical system.

The overall objective of SAUNA is an integrated framework for safety assessment and transparent safety demonstration. Both the regulator and utilities, as well as system suppliers and contractors, would benefit from cost-effective and timely licensing and implementation of investments in new builds and upgrades. The SAUNA project aims to contribute to this goal in the following ways:

- On the level of fundamental safety principles, SAUNA reviews recent trends in regulatory policies, standardisation and research on the design of complex, safety-critical systems for a better understanding of various aspects of the overall nuclear power plant safety. The resulting literature reviews, roadmaps and conference papers provide insights to necessary improvements in national design and regulatory practices and to the needs for further research, especially within the SAFIR2018 programme.
- For practical design and licensing work, SAUNA takes the good practices of Systems Engineering and project management as the starting point and adapts them to the needs of the nuclear domain. Progressing from the clarification of current terminology
towards more formal information models SAUNA promotes future model and
database oriented design practices as a complement to traditional documents. The
expected results on this level include shared reference models for various systems
engineering and regulatory processes as well as documents and data models for
expressing the information exchanged between the parties involved in design and
licensing. Process assessment methods are developed to evaluate the quality of the
systems engineering processes.

- Within the framework of Systems Engineering and principles of nuclear safety, the
  SAUNA project focuses on safety demonstration practices considering both the ways
to represent the safety claims, arguments and evidences and the processes of safety
case development and licensing. Safety assessment and justification is, however,
seen together with the design activities that produce a significant part of the claims
and evidences needed for safety demonstration. The results, primarily in the form of
guidance and (optionally) tools for safety case development and licensing, can be
used during new build and upgrade projects and for periodic safety reviews of
existing plants. The structured representation and development approach are
expected to make the design and licensing of power plant automation smoother and
more cost-effective, both for the regulator and the utilities.

- The claims presented in a safety demonstration are typically derived from regulations,
standards, specific system requirements and other points of reference like “best
practices in the domain”. For collecting a comprehensive set of evidences assessors
need a multitude of analysis methods. The objective of SAUNA is to enhance and
integrate existing methods and to develop new assessment methods where needed
for a good coverage in the safety case. While each method focuses on its specific
aspect of a system or its life-cycle, they all should provide useful inputs to the safety
demonstration and, where practical, apply the same claim-argument-evidence logic in
their working process and documentation of the observations. The results are
documented in the form of research reports and manuals that can be used by the
developers themselves and by independent assessors.

- The production and management of all the information needed for design and
licensing is not possible if performed manually. Therefore, SAUNA also intends to
develop software tools where the analysis of practical needs and implementation
options make it reasonable. Examples of potential areas for tool development are
management and exchange of design data (e.g. requirements, traceability, system
configuration and versions, PRA), analysis methods (process assessment, model
checking), safety case development and documentation, and utility-regulator
communication (e.g. issue management systems).

In order to limit the scope and the required efforts SAUNA focuses on design and licensing
issues related to power plant operations in normal, low-power and accident situations. This
means that I&C and information systems, control room(s), human operators and emergency
support personnel are in the centre.
2.1.6 GENXFIN - Safety of new reactor technologies

The main objective of the GENXFIN network is to improve scientific and technologic expertise in the field of innovative nuclear energy technologies. The knowledge is needed nationally to enable future nuclear reactors being deployed in a reasonably near future in Finland. Also international collaboration with ESNII, EERA, IAEA INPRO (Innovative Nuclear Reactors and Fuel Cycles), OECD/NEA CSNI (Economic Co-operation and Development / Nuclear Energy Agency the Committee on the Safety of Nuclear Installations) and other global forums such as GIF, is needed. The mission is not only to create enabling knowledge pool in Finnish needs, but also enable new business activities for the Finnish industry through enhanced technology transfer, innovative process development, and materials engineering. Also the safety authorities benefit from the outcomes of this project since they have the possibility to follow and steer the development work of the future reactor technologies. The activities in the programme will cover scientific, technological and industrial goals. Research & education organisations, safety authorities, manufacturing industry and power companies as well as ministries and other associated organisations are participating in the network.

The project consists of three work packages:

- WP1: Safety features of SMRs
- WP2: International co-operation
- WP3: Project management.

GENXFIN will follow developments and coordinate national projects related with new reactor technologies, focusing mainly on SMRs (Small and Modular Reactors) but including also Gen4 and CHP (Combined Heat & Power) topics. Additionally, combined technical-economic issues will be covered through case studies in the coming years. Load following and grid development related topics with new kinds of electricity generating systems as well as advanced fuel cycles will be followed within the project. The objective of the GEN4FIN project has been to enhance national expertise in science and technology of nuclear reactors. The project has had international cooperation within GIF, EERA (European Energy
Research Alliance), IAEA (International Atomic Energy Agency) and ESNII (European Sustainable Nuclear Industrial Initiative). This type of international cooperation is meant to continue within GENXFIN project in 2017 focusing mainly on the different IAEA activities (through consultant and technical meetings) as well as INPRO, and also WNA (World Nuclear Association) Working Group CORDEL (Cooperation in Reactor Design Evaluation and Licensing) and GIF SCWR (Supercritical Water Reactor) M&C (Materials & Chemistry) Project Management Board (PMB).

Figure 2.1.6.1. Small and modular reactor (SMR), in figure the SMART reactor (IAEA, 2012)

2.1.7 ESSI – Electric systems and safety in Finnish NPP

The objectives of research in the project are to examine the possible common cause fault impacts of OPC (Open phase condition) and large lightning strikes in Finnish NPP electrical systems. Also the risks of adaptive operation of NPP in load following mode will be examined. Analysis is made about following expected results:

OPC: severity of unbalance in different possible open phase situations is analysed. The consequences of possible voltage unbalances are assessed from different electrical system components point of view, and criticality of different OPC cases is analysed with regard the possible risks and time required from mitigation actions. An expected result will be how the operating plants have been prepared to OPC.

Lightning overvoltages: Voltage stresses caused by large lightning currents in Finnish NPP electric systems is analysed for both surges entering via transmission grid and through lightning strikes into the grounding system. The adequacy of overvoltage protection is assessed and possible improvements are suggested for the surge arrester sizing and location, as well as for the grounding arrangements of the electrical, automation and instrumentation systems.
Concerning the adaptive operation of NPP, the objective is to estimate requirements, technological limits and risks of adaptive control in today’s nuclear power plants with regard to electrical systems in order to avoid the increase of disturbances in power plant. The objective in 2017 is to obtain the minimum requirements for the manoeuvrability capabilities of Finnish nuclear power plants.

The ESSI project is divided into three work packages each having a planned duration of two years (see Figure 2.1.7.1). For each project year, a set of tasks will be defined with a clear scope and deliverables. While individual tasks may be short and their titles may be changed from the first year to the second one, they work on longer-lived topics within the work packages.

Figure 2.1.7.1. Tasks planned for 2017 in work packages 1 to 3 and their further planned continuation in the year 2018.
2.2 Reactor safety

In 2017 the research area “Reactor safety” includes 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

The objective of the project is to develop safety analysis methods which benefit the safe and sustainable use of nuclear energy in Finland. The capability of simulation tools in use, including integral codes and several specialised programmes to model phenomena related to severe accidents will be assessed. If needed, the codes and methods will be further improved in collaboration with colleagues around the world. Reinforcing international networks will bring the most recent relevant knowledge to the use of Finnish nuclear community. The objective is not only to follow the international actions but to adopt the latest information to Finnish context.

The project consists of four research work packages:

- Progress of severe accidents (WP1)
- Core melt management (WP2)
- Containment phenomena (WP3)
- Environmental consequences (WP4).

The main outcomes of the project are (1) comprehensively validated simulations tools available for the assessment of severe accident scenarios for the needs of the Finnish NPPs, (2) trained experts who can use these tools and have in-depth understanding on the complex physical phenomena and (3) significant reduction of uncertainties in the key processes that determine the consequences of a severe accident.

The results of the project will be published in the form of articles in scientific journals and conferences and as theses and dissertations of undergraduate and doctoral students. Thus, the high scientific quality of the results is ensured. At the end of this project, Finland will be able to more reliably analyse severe accident scenarios in our current and future NPPs.

In 2017 the MELCOR model of one of the units will be updated, using newly available plant data. A poster about the Fukushima calculations will be prepared for the ERMSAR conference. The BSAF-2 project meetings will be participated.
The effect of RPV breaking mode on dynamic pressure loads on cavity wall induced by steam explosions will be analysed with MC3D. The focus will be on evaluating the effect of break location and size. The cases will be carefully assessed to correspond to realistic conditions. This task is a part of the NKS-SPARC project.

Some of the pool scrubbing experiments made in the SAFIR2018 CATFIS project will be simulated with ASTEC and MELCOR. The focus is in evaluating the code capabilities to simulate not only the decontamination factor but also iodine liquid phase chemistry. Especially on the effect of pH there is a lack of validation data. A paper on the effect of pH control on iodine behaviour will be prepared for the ERMSAR conference.

Related to emergency preparedness in cases of radioactive release, VALMA will be augmented with the calculation of acute and late health effects of radiation doses to increase also its competence to assist in radiation protection and in level 3 PSA. The probability distributions of radiation-induced health effects at various distances and over various time periods will be produced. The results will be compared to some corresponding ARANO analyses. VALMA results are expected to be more reliable since the code utilizes realistic weather data.

2.2.2 CATFIS - Chemistry and transport of fission products

The main objective in CATFIS project (2015-2018) is to find out the mechanisms how gaseous fission products, especially iodine and ruthenium, are released from primary circuit into containment. In addition, the behaviour of iodine in the containment gas phase will be studied. Another objective is to find out the effect of radiation on the formation of air radiolysis products and their impact on the fission product speciation. Also the retention of fission products in the containment pool or FCVS scrubber due to the pool scrubbing phenomenon will be investigated. The data gathered will be utilized to develop mathematical models on fission product transport. The international collaboration will be further strengthened through the NKS-R collaboration with Chalmers University of Technology and through a direct collaboration with JAEA and IRSN. The international collaboration will be carried out by participating in OECD/NEA STEM-2 and BIP-3 programs and also in NUGENIA TA2.4 area international projects.

Figure 2.2.1.1. Dose rates in bulk gas phase inside containment (ASTEC and NRC method comparison).
The project is divided into five workpackages in 2017. The first workpackage is focused on the primary circuit chemistry of iodine. General objective of this work is to identify if chemical reactions and revaporation process at the primary circuit surfaces may have a significant effect on the physical and chemical form and on the transport of fission products. In the proposed work, the impact of initial Mo/Cs molar ratios on the formation of gaseous iodine in the circuit will be examined. In PWR fuel, the molar inventory of molybdenum is higher than the one of caesium by a factor of 1.5 to 2. Nevertheless, the Mo/Cs molar ratio in the primary circuit is highly dependent on the accident sequence and can vary over a larger spectrum during the transient. The precursor samples will be exposed to Ar/Air and Ar/H2O atmospheres at 650°C and the reaction products will be analysed in detail. The studies will give more information on the formation of the caesium molybdates and on the difference of the fraction of gaseous iodine released according to the initial Mo/Cs ratio. The chemical reactions of fission product deposits taking place on the primary circuit surfaces are not considered in the current SA analysis codes, although the deposits can act as a significant source of volatile iodine especially at the late phase of accident when e.g. thermal-hydraulic conditions are changing. By the end of 2018, the objective is to utilize the previous and new experimental data in developing models for the release of gaseous iodine from the FP deposits on RCS surfaces and for the subsequent iodine transport, as the phenomena has been studied in TRAFI and CATFIS projects for several years. It will take place as a visit at IRSN and as collaboration with their personnel to implement the models into the ASTEC/SOPHAEROS module. The collaboration with IRSN will be more active than before and several publications are expected to be finalized from this task. These new models and code developments will enable the consideration of phenomena, which were previously poorly-known, in the analysis of severe accident.

The formation of nitric acid by beta radiation in the containment gas phase and sump will be studied in the second workpackage. Previous studies on nitric acid formation have been conducted with gamma radiation. As mentioned above, in a severe accident the most important source of ionizing radiation is beta radiation. However, the rate of nitric acid formation due to beta radiation may be a lot higher than expected. The objective is to find out the formation rate of HNO3 by beta radiation (e.g. P-32) in humid air or nitrogen atmosphere. The concentration of water, oxygen/nitrogen ratio, dose rate and radiation dose will be varied. The results will be compared with the previous tests conducted with gamma radiation. The understanding on these objectives has increased due to experiments in 2015-2016. By the end of 2018, the formation rate will be defined in a temperature range from 20 to 120 °C. Another objective is to extend the experimental work to cover mixtures of water pool and air or N2 atmosphere as well, in order to gather information on the resulting HNO3 content in the water pool. The radiation source can be located in the gas phase or in the liquid phase. These studies will also give information on the formation of air radiolysis products in general and on their thermodynamic equilibrium with other species. The effect of the formed HNO3 on the sump acidification and on the possible release of iodine from the sump will also be considered, especially in the third workpackage (see below). This kind of information is needed to enhance the nuclear safety. The results will be summarized in a scientific publication, which will be part of the PhD thesis of Teemu Kärkelä (VTT).

The third workpackage is focused on the development of ChemPool software. The objective is to improve the chemistry model. That will include the updating of nitric acid formation rate to notice the impact of beta radiation as well. The data for it will be received from the experiments of WP2. Secondly, the chemistry model of Chempool will be updated continuously during the project in 2015-2018. The third objective is to recalculate Olkiluoto and Lovisa severe accident scenarios with the updated chemistry model and HNO3 formation rate. This task is important in the course of project, since it will give an overall view on the effect of various phenomena to pool pH and to speciation of iodine concerning especially Finnish NPPs and thus this knowledge will increase the nuclear safety. The results will be summarized in a scientific publication.
The objective in the fourth workpackage is to investigate the retention of fission products due to the *pool scrubbing* phenomenon. It can take place e.g. in the containment pool or FCVS scrubber. The objective is to focus on the long-term behaviour of the pool as a trap for the fission products. As a result, the trapping efficiency of the pool when it has filled-up with fission products and other structural materials will be found out. Another objective is to investigate the behaviour of organic iodine in the pool. As organic iodine is known to be difficult to trap, the effect of its volatile nature on the release from the pool will be verified. As a third objective, also the retention of aerosol particles will be studied. The main emphasis in this workpackage is to understand how the pool behaves in a long-term. And as another main factor of this WP, the pool temperature will be varied from 20 to 100 °C in the experiments. Also the pool chemistry will be as realistic as possible in the experiments corresponding to Finnish NPPs. It is possible that when the pool with some oxygen content is exposed to heat and radiation it results in a change in the alkaline chemistry of the pool, which may decrease the retention of fission products in the pool significantly. This needs to be studied with experiments. Additionally, several other pool parameters will be varied in the experiments, see Chapter 2. The work will be initiated in 2017 with a focus on the needs of Finnish nuclear power plants. VTT will get technical advice from PSI (Switzerland), which has performed pool scrubbing experiments previously and thus PSI has also knowledge on the hydrodynamics of the pool. Additionally, VTT participates in the NUGENIA TA2.4 area project IPRESCA, which begins in 2017. The IPRESCA project summarizes the current knowledge on the pool scrubbing phenomenon internationally. The experimental data of WP4 will be also utilized in the SAFIR RG2 CASA project, in which the experiments will be simulated with the MELCOR and ASTEC codes and the calculation results of the codes will be compared. The gathered knowledge in the VTT’s pool scrubbing experiments and modelling will also be shared through OECD/NEA THAI-3 program. The follow-up of THAI-3 is being performed in the SAFIR2018 RG2 CASA project. The publications of the results from this task are part of the PhD thesis of Teemu Kärkelä (VTT).

The *follow-up of OECD/NEA STEM-2 (Source Term Evaluation and Mitigation Issues)* and *OECD/NEA BIP-3 (Behaviour of Iodine)* programmes, which both began in January 2016, will be continued in the fifth workpackage. The four-year STEM-2 programme is focused on the transport of ruthenium in the primary circuit conditions and on the reactions of particulate iodine on the painted and metal containment surfaces. The three-year BIP-3 programme is focused on the behaviour of gaseous inorganic and organic iodine on the painted containment surfaces, especially the adsorption and desorption phenomena. The specific project plans of both programmes were distributed to SAFIR2018 RG2 members in summer 2015. Due to the VTT’s unique experiments on the Ru chemistry in RCS (SAFIR2014 and SAFIR2018 programs), VTT’s technical advice and expertise has been asked in the planning of experiments in OECD/NEA STEM-2 program. Another area of VTT’s expertise will be needed on tests related to the release of gaseous iodine from iodine containing aerosol deposits on containment surfaces. As described above, VTT has previously successfully performed similar experiments, under SAFIR2014 program as well, and the results have been taken into consideration in the planning of STEM-2. Both OECD programmes produce valuable data on the behaviour of fission products, especially iodine and ruthenium, which is needed in the estimation of possible source term.
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.

The overall objective of the project is to improve the state of validation of two mutually-independent safety analysis tools, Apros and TRACE, through a systematic and rigorous approach to the validation process, and also promoting this kind of approach to the validation process. The process enhances the expertise in thermal hydraulic area of Generation II and III LWR reactors and includes as an essential part training of new experts to this relevant area of reactor safety. While the main effort is carried out using Apros, as it has higher national interest as a self-developed independent and versatile safety analysis tool, TRACE is also used in analyses of new experiments and in code-to-code comparisons with Apros.
In particular, the project answers in its own part in multiple research goals and aspects highlighted within the SAFIR2018 framework plan:

- Validation of independent thermal-hydraulic calculation codes, and especially modelling of new types of safety systems has been deemed as still requiring more resources
- Validation of all software used in computational nuclear safety analyses is encouraged to be developed to a more systematic and extensive direction
- Use of OECD/NEA’s thermal-hydraulic validation matrices, the content of which have been last time updated in 2010, is encouraged
- Participation in computational analyses of reactor safety experiments carried out in international research projects is encouraged to get full benefit from these projects
- The project promotes international networking and cooperation through participation in international research projects such as the CAMP programme and projects coordinated by OECD NEA. Experimental data produced in these projects is also essential input for the work performed in this project
- The project supports, in its own part, experimental activities carried out at Lappeenranta University of Technology. Also experiments carried out at LUT provide useful input for this project
- The project has an important educational aspect that aims at competence transfer from an older generation of scientists to younger researchers.

COVA is divided into four work packages (WP’s): Validation matrices (WP1), Analyses of new experiments (WP2), Management and international cooperation (WP3) and Participation fees (WP4). The actual research work dealing with analysis tool validation is carried out in the first two work packages, with WP1 concentrating on the fundamental aspects of the validation work with Apros, and WP2 in application of Apros and TRACE to validation using primarily integral-scale experiments with proper quantification of output uncertainties.
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 will be gathered on

- effect of SRV spargers, RHR nozzles, strainers and blowdown pipes on mixing and stratification of the pool;
- 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;
- formation of liquid films on the vessel wall and blowdown pipe due to spray operation and their effect on heat transfer and local condensation and heat flux to the pool;
- earlier suppression pool test results concerning blowdown pipe with a collar, parallel blowdown pipes and transparent/poorly conducting blowdown pipes.

To achieve the objectives a combined experimental/analytical/computational program will be carried out. LUT will create an experiment 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 experiment database for the development, improvement and validation of numerical simulation models. Also analytical support will be provided for the experimental part by pre- and post-calculations of the experiments.

Expected results from 2017:

- Wetwell spray experiments will reveal if mixing of a thermally stratified pool with the help of spray injection from above is successful.
The sparger test series in PPOOLEX will be concluded with a final experiment and closures for the EMS model development work for spargers will be provided.

- Effective momentum induced by steam injection through a SRV sparger will be directly measured in a small scale separated effect test facility.

- Survey of noncondensible gas dissolution/release dynamics modelling will be provided.

The main benefit of the project will come through improved and validated calculation models of CFD and system codes used for nuclear safety analysis. 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. The research results can be used by the power companies, nuclear safety authorities and research organizations.

Figure 2.2.4.1. An example of chugging bubble at 14s of simulation.

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

The main aim of the project is to ensure the operation of safety related systems or the efficiency of the procedures in accident and transient situations of nuclear power plants. An integral test facility, such as PWR PACTEL, offers a good possibility to carry out tests which supplement test campaigns in the other facilities (PKL) or make independent tests to study phenomena relevant to the safety of nuclear power plants (pump trip etc.) As a result, counterpart like tests give information of parameter effects such as a smaller scaling ratio or a higher pressure level (PWR PACTEL/PKL) when certain operator actions or system activation set points are used.
The limiting factors of the operation of passive heat removal loops were surveyed and the PHRS-C system of AES-2006 design was selected in 2015 to be studied experimentally. The ultimate goal is to identify physical mechanisms that can reduce performance or prevent the functioning of the loop, to help recognizing conditions in which the functioning of the system could be endangered and to suggest ways assuring the operation. The design work of the facility began in 2016.

The expected results of the INTEGRA project are:

- LUT participates in the OECD/NEA PKL Phase 4 project with the PWR PACTEL facility,
- The PWR PACTEL experiments to study the flow reversal due to a pump trip,
- A system to investigate the fundamentals of the PHRS-C passive system of AES-2006 design.

![Diagram of passive heat removal test facility in LUT](image)

Figure 2.2.5.1. Tentative structure of the passive heat removal test facility in LUT.

2.2.6 KATVE - Nuclear criticality and safety analyses preparedness at VTT

The objective of the project is to improve the preparedness to perform safety analyses in the field close to reactor physics as well as raise the knowledge of the current researchers and educate new experts to the field. The main analyses are in the field of criticality safety and radiation shielding. Additional analyses are activation of structural materials and neutron dosimetry. Moreover, source terms are often needed for analyses also outside the scope of
this project. These source terms comprise nuclide inventories and decay heat. In this area, the ability to build a complete analyses chain from the heat source through the CFD calculation for heat transfer to the integrity of fuel is an important outcome.

The objective of the criticality safety work is to have an appropriately validated calculation system that has the possibility to take into account also the burnup credit. In addition to the tools, also the knowledge on how to perform a criticality safety analysis is revived. In the four years of the programme, the criticality validation package should be finalised and depletion validation of the calculation tools should be in good progress. The validation package and burnup validation reports are concrete results of the project. In addition, the calculation tools are developed to take into account the burnup credit in an easy and manageable way.

Serpent is developed into a practical simulation tool for criticality safety, radiation shielding and other applications involved in the project. Once the work is completed, the code is capable of performing reliable criticality safety analyses, producing source terms for radioactive inventory and decay heat calculations, as well as calculating neutron and gamma dose rates in complex geometries using the best available knowledge on interaction physics and state-of-the-art variance reduction techniques. The use of the continuous-energy Monte Carlo method with unstructured mesh-based geometry types and CFD code coupling constitute a one-of-a-kind calculation scheme for the safety analysis of spent fuel storage facilities.

Shielding analyses involve use of lighter codes like Microshield, in addition to the heavy Monte Carlo calculation. An important objective of this project is to learn to use the code with the source terms calculated with Serpent and ORIGEN-S.

The work in the project has been devided into four work packages according to the analysis under consideration:

- WP1: Criticality safety
- WP2: Radiation shielding
- WP3: Source terms and activation
- WP4: Heat transfer and fuel integrity.

The emphasis in the work is in developing methods and tools for making analyses in the areas chosen for this project. The tools and the knowhow will ensure a readily available capability to perform criticality and safety analyses for the authority and the utilities.

Figure 2.2.6.1 MCNP silo and source geometry.
2.2.7 MONSOON - Development of a Monte Carlo based calculation sequence for reactor core safety analyses

The project continues the development of the Serpent Monte Carlo code, started in 2004, and carried out within the previous SAFIR programmes. Compared to the KÄÄRME project in SAFIR2014 the work is clearly more focused on spatial homogenization, and 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 improved methodologies are thoroughly validated and put to practice in the calculation schemes used at VTT for the safety analyses of Finnish power reactors.

The Serpent code, together with the fuel cycle and transient simulator codes developed at VTT and STUK, form a complete and independent calculation sequence for the safety analyses of Finnish power reactors. The continuous-energy Monte Carlo method provides a novel approach to spatial homogenization, enabling more rigorous modelling of conventional, as well as advanced fuel types and axially heterogeneous cores.

The work supports other proposed projects in SAFIR2018, in which Serpent is involved as a calculation tool for various applications, in particular:

- **SADE (VTT)** Serpent is used for group constant generation for TRAB3D and HEXTRAN transient reactor analysis codes and as a reference for analyzing the accuracy of the codes and the results of improvements implemented into the codes.

- **PANCHO (VTT)** – The FINIX fuel behavior code developed in the project is internally coupled to Serpent.

- **KATVE (VTT)** – Serpent is used for criticality safety analyses and calculating radioactive inventory and decay heat source terms. The project also involves development of photon transport mode in Serpent for the purpose of radiation shielding applications in spent fuel storage facilities.

The project plan for MONSOON 2017 continues the work related to VVER reactors, in close in-kind collaboration with the Helmholtz-Zentrum Dresden-Rossendorf (HZDR). Validation studies are also extended from PWR to BWR applications, which essentially means covering a much larger state-point matrix in group constant generation.

Other topics, carried out in parallel with group constant generation, include accounting for the effects of fuel temperature feedback on assembly burnup calculations, and novel methods for cross section parametrization. The work began in 2015 with a study involving coupled Serpent-ENIGMA burnup calculations, providing realistic assembly- and pin-wise temperature profiles over the irradiation cycle.

Since close collaboration with the large international Serpent user community and feedback from expert users has proven extremely valuable for the development work, part of the project is devoted to maintaining close contacts with the users.
2.2.8 NURESA - Development and validation of CFD methods for nuclear reactor safety assessment

The importance of the validation of the codes is emphasized in the SAFIR2018 Framework Plan [1]. No comprehensive validation matrix for CFD codes exists, but some recommendations on the validation have been given by OECD working groups. The procedures for the validation of CFD tools have to be developed so that CFD analysis can be used in licensing calculations of NPPs. Therefore, CFD grade validation data is needed that can be obtained, for instance, with Particle Image Velocimetry, tomography or wire-mesh sensors. The nationally important experimental data should be listed and the missing experiments should be performed in national or international projects. The validation of the numerical tools for analysing Fluid-Structure Interaction (FSI) should be performed.

The NURESA project proposal consists of four year Work Packages, where CFD methods are developed and validated for the identified most important topics in NRS assessment. In Work Package 1 (WP1), international single-phase mixing benchmarks are participated. In WP2, PPOOLEX spray experiments are modelled with CFD codes in co-operation with Swedish partners. In WP3, CFD models for DNB and dry-out are developed for Open-FOAM code in co-operation with international partners. In WP4, CFD-Apros simulations of NPPs are performed to validate the coupling of CFD and system codes. Coordination of the project is done in WP5.

The developed and validated CFD methods are used by the regulator, utilities and research organizations in NRS assessment. In addition to the validation of the CFD methods, the uncertainty of the results is assessed in international benchmark calculations. The information on uncertainties is essential in using CFD in NRS analysis.

The benchmark calculations performed in WP1 provide validation of the CFD methods in the calculation of mixing and stratification. In addition, information on the uncertainties of the results is obtained, when the results of several international partners are compared to experimental results.

The modelling of PPOOLEX experiments in WP2 provides improved understanding on the pressure suppression function of the BWR containment. New models for condensation and evaporation of spray droplets and liquid films can also be used in other NRS problems, such as in modelling of pressurizers of PWRs.
The subcooled boiling models of OpenFOAM that are developed and validated in WP3 can be used after the second year of the project for NRS assessment. A model for DNB is available at the end of the four year project. A main result of WP3 is a publicly available, transparent and efficient open source CFD simulation tool for nuclear safety. In addition, a related national toolset is formed with application specific closure models and best practices. This will provide a common software platform for national and international co-operation in the field of nuclear safety related CFD.

The validation of coupled CFD-Apros calculations in WP4 provides a new analysis tool for NRS assessment. The coupled calculations make possible to analyse large systems with three-dimensional components.

![Velocity magnitude of air in the vertical center plane through the nozzle.](image)

### 2.2.9 PANCHO - Physics and chemistry of nuclear fuel

The project 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 the chemistry of the fuel pellet and the mechanical response of the cladding.

As the nuclear fuel provides the first physical safety barriers against the spread of radionuclides, understanding the fuel behaviour during accidents is vital. This information must be gained through experiments, as the phenomena encountered are complex. A large part of PANCHO focuses on international programmes either performing experiments or distributing the data and tools for and experience on fuel behaviour analyses.

PANCHO investigates several nuclear fuel related issues with an interdisciplinary approach combining theoretical and experimental investigations. This work supports both the in-depth understanding of safety-related phenomena and communication across disciplines. The research will support several dissertations, and thus it will be reported in scientific journals and conferences to ensure the high quality and visibility of the work. As such the project is both relevant to nuclear safety and aims for high scientific quality.

The work in PANCHO is divided into four work packages:

- **WP1: Computational framework**
  - FINIX is a general purpose fuel behaviour module for thermal and mechanical fuel behaviour in multi-physics simulations, and has been integrated into VTT’s Serpent 2 reactor physics code and reactor dynamics codes. In PANCHO, further development and validation of the FINIX fuel behaviour module is a major goal during 2015-2018. The most important phenomena
that are not yet modelled in FINIX include cladding creep, cladding oxidation, cladding ballooning, cladding burst and fission gas release. In 2017 the focus will be in modelling as many of these phenomena as possible and implementing the models in FINIX.

- WP2: Integral fuel behaviour
  - This work package consists of three tasks: the LOCA and RIA performance of the fuel and the Halden in-kind work.
  - A coupling of an external thermal hydraulics code with SCANAIR was done as an in-kind work for 2015. The VTT in-house general thermal hydraulics code GENFLO was chosen for the coupling. The purpose of this work was to make possible the modelling of BWR thermal hydraulic conditions with SCANAIR, as this was not possible with the current models in the code. Validation simulations on the coupling were done in 2016. A journal manuscript of the code coupling and validation was prepared, and in 2017 the work performed with RIA and LOCA analysis will be collated to Asko Arkoma’s DSc dissertation.

- WP3: Separate effects
  - Leaching behaviour of fuel materials
  - Cladding mechanical response and load follow

- WP4: Management and international co-operation.

Figure 2.2.9.1. Cladding temperature evolutions plotted from various radial nodes from SCANAIR and GENFLO. The axial node in all curves is no. 5. Steady-state power 15 kW/m. (Arkoma, 2016)
2.2.10 SADE - Safety analyses for dynamical events

VTT has been at the forefront in the development of the coupled reactor dynamics codes during the last decades, when coupled neutronics-thermal hydraulics codes HEXTRAN and TRAB3D were developed. Aim of the project is to model transients and accidents in such a way, that we can give more reliable answers to the safety requirements set in the YVL guides. The main idea is to improve VTT’s modelling capabilities by routine coupled use of the CFD-type thermal-hydraulics solver PORFLO and the reactor dynamics codes HEXTRAN and TRAB3D. New submodels for wall friction and mixing are required especially for two-phase conditions. Also the neutronics modelling needs to be more detailed and the whole safety analyses methodology revised to get the full benefit on the 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. Own code and in-depth understanding of it enables the best possible expertise on safety analyses.

The developed computational tool set of coupled neutronics, system codes and real 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 several transients and accidents of real interest have been calculated. First cases to be calculated are a re-connection of an isolated circulation loop filled with low-temperature coolant and a main steam line break (MSLB), both for a VVER plant. After these the objective is to calculate other cases in which three-dimensional phenomena are significant, including also cases in which two-phase modelling is required:

- asymmetric flow transients such as pump transients
- asymmetric reactivity transients such as control rod ejection (CRE), control rod withdrawal (CRW)
- sudden changes in coolant conditions such as pressure transients in BWRs, boron dilution, propagation of a cold water front
- Failures in operation or protection such as loss of offsite power (LOOP), load rejection, ATWS
- BWR stability

A further objective is to calculate also transients, which cannot yet be modelled with existing tools

- Mechanical interaction of flow and fuel assemblies: fuel rod bowing, lift off
- Blocked flow channels.

In 2017-2018, the development of the two-way coupled HEXTRAN-PORFLO-SMABRE computation system will continue. The validity of the new coupled modelling framework in whole-plant safety analyses will be examined. The coupling system will be improved, streamlined and completed for different kind of coupled analyses to be performed. The simulations with the coupled codes will be published in papers, conferences and AER symposiums.

In 2017, two transient computations will be performed with the two-way coupled HEXTRAN-PORFLO-SMABRE code system. A new mesh covering the whole RPV will also be created for VVER-440 in 2017. In addition, the VVER-1000 meshes will be extended to cover also the reactor head.
2.2.11 USVA - Uncertainty and sensitivity analyses for reactor safety

The project builds on the existing expertise in uncertainty and sensitivity analyses at VTT and Aalto University and merges the on-going research activities under one project. In addition, USVA promotes activities at the interfaces of the different disciplines in 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 whole reactor safety field.

USVA approaches the general goal on several fronts. The problem of performing a complete analysis of the whole coupled code system involving uncertainties from various stages of the calculation sequence and from the different physical sources is simply too complex. For this reason, the problem is analysed in parts. These parts form the concrete objectives of the project.

During the four-year period, USVA is expected to advance the knowledge in the analysis of multi-physics coupled calculations by studying both steady state and transient behavior of the reactor. Methods for propagating uncertainties through calculation sequences will be developed, with applications in reactor dynamics calculations and in fuel rod failure analyses. The methods will be implemented in safety analysis codes and tools at VTT. These objectives are also supported by the knowledge gained from individual code and accident scenario analyses, and by the methods developed for input uncertainty estimation. In addition, USVA aims at developing an effective practice for uncertainty and sensitivity analyses of multi-physics systems, such as combined neutronics and thermal hydraulics or fuel behavior simulations. In addition, the most influencial uncertainty groups will be identified in such simulations.

The research work packages and tasks of the USVA project in 2017 are:

- Methods and analyses (WP1)
  - Analysis of rod failures in LB-LOCA
- Methodology for determining input uncertainties
- Collision history-based GPT capabilities for Serpent

- Multi-physics and the calculation chain (WP2)
  - Coupled calculations with fuel performance and reactor dynamics codes.

Figure 2.3.11.1 Neural network predictions of the rods that were simulated by FRAPTRAN-GENFLO to (a) fail and (b) survive. All the two-cycle rods in the reactor are included in this test set except those 1000 used for training the network. The data is split into 200 bins.
2.3 Structural safety and materials

In 2017 the research area “Structural safety and materials” includes 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

An aircraft impact on safety related structures, in spite of its low probability, has for a long time been recognized as a relevant loading case, especially in designing plants to areas with heavy air traffic. Structural analyses of related phenomena require nonlinear numerical analysis methods. In order for the results of these numerical analyses to be reliable, the applicability of the used methods and models should be validated by experimental results and analytical methods.

The main aim of this work is to develop and take in use improved methods and modelling techniques which are validated against experimental results. Experimental data on nonlinear dynamic behaviour of reinforced concrete structures loaded by hard and deformable missiles has been obtained at VTT within IMPACT projects (Phases 1, 2 and 3) already for over 10 years. The next project, IMPACT 4, is now under planning. International partners from previous projects are willing to continue the cooperation. The pre- and post-studies related to the IMPACT project will be carried out within the ERNEST project. Some dedicated tailored tests for domestic purposes will be undertaken within the ERNEST project.

Models and methods for assessing structural integrity of impact loaded reinforced concrete structures are developed and validated utilising experimental data. In practice, post calculations of impact tests are an important way to identify needed development work.

Work packages and tasks of the project are:

- Experimental studies (WP1)
  - Punching behaviour test with two consecutive slabs
- Modelling of nonlinear behaviour of RC structures (WP2)
  - Concrete material model development
  - Pre- and post-analyses for IMPACT and ERNEST tests
  - Vibration propagation and damping.
2.3.2 FIRED - Fire risk evaluation and Defence-in-Depth

A significant proportion of the overall core damage risk in nuclear power plants (NPP) is associated with internal fires. In addition, a fire on NPP can cause large financial losses even if the risk to the reactor safety is small. Therefore the possible initiating event scenarios and the operation of defence-in-depth after ignition are important topics in the research of nuclear safety. The computational tools that are used for assessing the fire risks have developed significantly over the last ten years: The deterministic analyses are now solely based on CFD and the probabilistic analyses using Monte Carlo simulation have been carried out.

FIRED-project will cover three main themes: fire risks of cables during the plant life cycle, fire Defence-In-Depth, and modelling tool development and validation. In addition, active participation to OECD PRISME 2-project will continue.

The main objective of the FIRED-project is to develop the tools for fire risk evaluation and create a new methodology for assessing the defence-in-depth in the context of fire safety. To meet this main objective, the following technical or detailed objectives are specified:

- Development of a pyrolysis modelling capability for the new flame retardants for predicting their performance in nuclear power plant fire scenarios
- Quantification of the ageing effect on modern cables through multi-scale experiments and modelling
- Development of a capability to predict the fire resistance of a barrier element with the Fire-CFD and – when necessary – the 3D-FEM tools. The fire resistance should include the aspect of load bearing (R), integrity (E) and insulation (I)
- Exploring the wider context and possible implementations of the Fire-defence-in-depth concept
Continuous development and maintainence of the fire modelling tools to meet the needs arising from the increasing community of end-users, to maintain the simulation competence, and to solve the found issues and problems of the software.

Participation in steering and utilization of the on-going OECD PRISME2 –project.

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.

![Figure 2.3.2.1. Modelling tools for fire research and engineering.](image)

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

The project 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.

The focus areas are as follows:

- 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 and transferability; beginning with a master’s thesis.
- WP4: Fatigue and crack growth caused by thermal loads, with emphasis on modelling, and including a licentiate thesis.

- WP5: Development of RI-ISI methodologies, including participation to ENIQ Task Group Risk (TGR) activities.

- WP6: Dynamic loading of NPP piping systems.

- WP7: Residual stress relaxation in BWR NPPs.

FOUND-project provides new experimental capabilities and more accurate computational lifetime and risk assessment applications for piping systems, BWR RPV internals and other NPP components. Through participation in international networks such as NUGENIA TA8 ENIQ TGR there is valuable international co-operation. The new experimental and analysis applications developed and tested in the project can later be used in tailored contract works.

![Figure 2.3.3.1. Contour plot of the effective stress intensity factor data used in the crack growth assessment. The crack growth path is shown with the red line.](image)

2.3.4 LOST - Long term operation aspects of structural integrity

The goal of the project is to develop through experimental and numerical methods more accurate structural safety assessment methods to the nuclear power plant (NPP) end users.

In WP1, Advanced structural integrity, the objective is to develop new advanced structural integrity methods to describe the ductile crack growth during a temperature transient accounting for temperature history effects and to develop a constraint, loading rate and crack growth adjusted modified advanced Master Curve methodology to deal with complex events related e.g. to leak before break (LBB) assessment. These advanced evaluation methods influence also the accuracy of RPVs structural integrity. This topic increases the knowledge of fast fracture in the upper shelf temperature, which is also required in the YVL guidelines.

The objective in WP2, Dissimilar metal welds, is an enhanced treatment of weld residual stresses in repaired DMWs, and utilisation of residual stresses in fracture assessment. Improvements in residual stress estimation increase the accuracy of the estimation of the remaining life.

Secondly, experimental and numerical investigations are used in WP2 to develop the calculation methods for crack driving force and fracture toughness in DMWs. The aim of the experimental investigations is to retrieve a deeper understanding of the fracture behaviour in DMWs interface regions, develop a method to characterise the near-interface zones (NIZ) of
DMWs and to produce data for the numerical investigations. The results of the experimental characterisation are used for calibration of FE models that are used to improve the analytical solutions of fracture toughness and to develop crack driving force solutions for NPP DMWs. These objectives impact the methods used to estimate the critical crack size. The results can be incorporated into the LBB analysis and leakage probability calculations in future projects. Results and mock-up from European collaboration project MULTIMETAL are exploited in WP2.

The 2017 plans of the LOST project are described in detail in the Appendix 1.

![Figure 2.3.4.1 Computed hoop stress [MPa] after nozzle/safe-end welding at room temperature (left) and after overlay welding (right). Only part of the model is shown. The reaction force at the cutting plane is also shown.](image)

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 due to impurity enrichment. Typical crevice liquid may be highly caustic with pHRT > 10 (NaOH) in addition to several other corrosive species causing pitting corrosion, denting and stress corrosion cracking.

The main objectives of the present study aim at developing knowledge and PWR/WWER secondary side water chemistry programs enabling

- replacement of hydrazine in the secondary side both during outage (in SG preservation) and power generation
- minimisation of magnetite formation in the feed water line
- minimisation of deposition of magnetite particles into SGs
- mitigation of corrosion phenomena in SG crevices related to deposition and impurity enrichment.
Specifically, as a result of the study the benefits of using film forming amines as passivating agents for carbon steel and inhibitor for lead assisted stress corrosion cracking (PbSCC) will be clarified. The role of boron, lithium and potassium on PWSCC of stainless steels and nickel base alloys in PWR/WWER primary water is investigated. The expected outcome is to improve the knowledge basis on which decisions on advanced secondary side water chemistries are made. These results will be used in plant life extension management programs.

Within EU, replacing hydrazine in the secondary side water chemistry is becoming a more important issue. Alternative approaches to the use of hydrazine during outages have already become an acute research issue.

The results of this project will be exploited when considering the use of different water chemistry alternatives. The results from studies of the effects of different amines and combination of amines as well as the results from mitigation of PbSCC can be applied in a longer run, starting from 2017. The end users are the plants (Loviisa 1, Loviisa 2, Olkiluoto 3 and Hanhikivi 1) and authority (STUK) in Finland as well as the nuclear community as a whole.

![Graph](image-url)

**Figure 2.3.5.1** Slow strain rate testing (SSRT) results for the two first test runs. Eoc = corrosion potential, i.e. test run without external polarisation.

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

The project THELMA, Thermal ageing and EAC research for plant life management, deals with nuclear materials behaviour in LWR environments with special focus on determination of thermal ageing in austenitic primary circuit materials (stainless steel weld and cast materials as well as Alloy 690 and Alloy 52 weld metal), the effect of irradiation on internals, the effect of environment on fatigue life and on precursors for environmentally assisted cracking initiation to be used for plant life management and failure analyses. In 2017 a new topic is introduced, i.e., investigations on the correlation between pressure vessel steel
microstructure and mechanical properties. Educating new experts in the field of nuclear materials is of high priority in the project.

Understanding and measuring the long-term effects of LWR environments on the characteristics of nuclear materials is essential for safe nuclear power plant operation. Materials are inherently subject to slow microstructural changes, i.e., thermal ageing, at LWR temperatures, and this will affect the properties of the materials. Thermal ageing changes the properties of the materials, and also increases their environmentally assisted cracking susceptibility. Knowledge is not only needed on thermally aged materials, but also on material characteristics of typical nuclear components in as-manufactured condition to enable prediction of their behaviour during the long-term operation. THELMA will address these issues.

The objectives of THELMA are to understand the underlying mechanisms and effects of thermal ageing, irradiation and nuclear environment on austenitic nuclear materials, i.e., austenitic stainless steel weld metals, cast stainless steel CF8M and nickel-based material Alloy 690. In 2017, investigations are started also on the effect of nuclear environment on pressure vessel steel weld metal with the objective of producing data on the correlation between microstructure and mechanical properties. The objective of investigation method development is to constantly improve our capabilities utilised in root cause analyses for our licensees.

The expected results of the THELMA project are:

- Determination of the activation energy for spinodal decomposition and G-phase formation in Type 316L weld metal and comparison to that of cast CF8M stainless steel.
- Determination of the changes in properties due to short-range ordering in Alloy 690.
- First steps towards best practises for surface quality in primary components.
- New experimental data and new guidelines for assessment of environmental fatigue damage to ensure safe operation of European nuclear power plants.
- Benchmarking of our capabilities to perform initiation testing in simulated LWR conditions.
- Data on the correlation between mechanical and microstructural characteristics of pressure vessel materials.
- Education of new nuclear materials experts
- Strengthened international co-operation and joint scientific publications

Assessment of thermal ageing propensities and mechanisms affecting thermal ageing and EAC of nuclear materials is needed as part of plant life management performed by the licensees. The safety authority can use the results and increased knowledge gained in THELMA in their work, securing safe operation of NPPs in Finland.

Potential safety issues connected to load following operation are highlighted in the update of the framework to the 2017 call. Load following will increase the amount of transients, which are known to be one of the main factor for environmentally assisted crack initiation. It will also increase EAC crack growth. These aspects can be investigated using the same methods for crack initiation as used in THELMA2016 in the MICRIN+ project, including transients. However, autoclave investigations are tedious and expensive, and to obtain quantitative data, a much bigger, multi-laboratory programme would be needed. These
aspects are therefore not included in the THELMA2017 proposal, but can be included in THELMA2018 if the MEACTOS project, which would support such investigations quite well, is approved. INCEFA+ supports the question on load following consequences.

![Figure 2.3.6.1 Post-test image of the wedge shaped specimen, showing the spray-on dot mask (black spots) and the strain field (blue to red, where red represents the highest degree of strain) in the sample.]

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

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-project includes of two work packages in 2017:

- NDE on primary component materials (WP1):
  - The propagation of ultrasound in austenitic SS and DMW welds
  - Assessing NDE reliability – experimental and model assisted POD for the nuclear industry

- NDT testing of concrete (WP2):
  - Preparation of the construction of the thick-walled reinforced concrete mock-up
  - Probability of Detection methodology applied to concrete NDE
  - Detailed design and construction of non-destructive evaluation and monitoring mock-up
The continuous development of the NDE methods for ISI is needed. In the WANDA project, this development will continue addressing the expressed needs of the NPPs. Main focus 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.

Additional objectives are:

- to analyse the differences in artificial defects and further verify the reliability of NDE simulations.
- to create an experimental POD in particular for nuclear application, e.g. for components in the primary circuit,
- to participate the international cooperation within U.S. Nuclear Regulatory Commission (NRC) PARENT (Program to Assess the Reliability of Emerging Nondestructive Techniques) and its follow-on program.
- to critically assess the NDT techniques and monitoring systems currently in use to fulfill the needs of NPP infrastructure evaluation in Finland.
- to develop guidelines for the use of NDE techniques in design and condition assessment, for the implementation of monitoring systems, and for performance based design and ageing management of the concrete infrastructure.

Further the goals of the project are to assess existing and new technologies such as NDT techniques for concrete examination and improve the power plants ISI techniques.

Furthermore, a mock-up of a full-scale concrete wall with artificial defects will be designed and built in the project. The purpose of the mock-up concrete wall is to allow for continuous long term testing and monitoring (greater than 20 years) which allows for different equipment to be assessed in a well-documented situation.

The results of WANDA project can be exploited by all the domestic NPP’s both operating and under construction and the information can be shared with foreign partners. NDE is one of the essential areas of research on the safety aspects of the NPP’s.

![Figure 2.3.7.1. Defect with TRS 2 technique at 40-70° SW from near side. Tip maximum at 62° angle.](image)
2.3.8 COMRADE - Condition monitoring, thermal and radiation degradation of polymers inside NPP containments

COMRADE is developed based on input from a feasibility study from Energiforsk AB an ongoing study ordered by STUK and through discussions between VTT Technical Research Centre of Finland, 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 radiation and synergistic effects between thermo-oxidative and irradiation degradation was identified. The plan is to divide the work into different steps, all with the aim of providing the power plant operators as well as regulators and polymer manufacturers with a deeper knowledge of the degradation of polymers and to develop methods for setting acceptance criteria of polymeric materials.

This project consists of three main objectives which include improving condition monitoring of polymeric components used inside containments, providing ageing data which is used in evaluation of acceptance criteria and providing tools for robust lifetime management for these components. To achieve these objectives three different work packages have been created which study polymer ageing in different perspectives.

- WP 1 focusing on method development of condition monitoring and implementation at NPPs
- WP 2 is a pre study to map how closed down power plants, such as for example Barsebäck, Oskarshamn 1, and materials taken from outages in running NPPs can be used to verify the method developed in WP1 and degradation process studied in WP3
- WP3 focusing on polymer ageing mechanisms and effects inside the NPP containment and ageing management of these components
- WP4 improves international cooperation on polymer ageing issues at NPPs between Nordic researchers, NPP operators and regulators.

The results gained from the project will allow regulators, power plant operators and polymer manufacturers to work with polymeric materials with greater knowledge concerning ageing phenomena and acceptance criterias. This will allow better monitoring of polymeric materials, life time prediction and to make sure the component is replaced at the correct time. This will also help estimating the status of a component before and during accident conditions. The project contains tasks for implementation in the work packages which means that after the implementation phase it is estimated that the knowledge, test method or other result can be used by the regulator, power plant operator and polymer manufacturer.

The 2017 plans of the COMRADE project are described in detail in the Appendix 1.
Figure 2.3.8.1. Lipalon cables after irradiation ageing at 28.6°C. From left to right: total absorbed dose 2.3 kGy, 23.3 kGy and 228 kGy, respectively. The used dose rate during all irradiations was 0.39 kGy/h
2.4 Research infrastructure

In 2017 the research area “Research infrastructure” includes three projects:

1. Development of thermal-hydraulic infrastructure at LUT (INFRAL)
2. JHR collaboration & Melodie follow-up (JHR)
3. Radiological laboratory commissioning 2017 (RADLAB).

2.4.1 INFRAL - Development of thermal-hydraulic infrastructure at LUT

The up-to-date experimental research infrastructure is essential for the modern nuclear safety analyses. The implementation of novel measuring techniques in the thermal-hydraulic experiments is needed for the validation of the Computational (Multi-)Fluid Dynamics (C(M)FD) methods. During the last decade, the use of CFD methods has become more common in the safety analyses of nuclear power plants. In order to rely on those analyses, one needs to have the credible validation of the method against experimental data from the CFD grade measurements. The reason for the growing popularity of CFD analyses is the complexity of many thermal-hydraulic phenomena that cannot be accurately predicted using the one-dimensional system codes. Since more complex phenomena also requires more advanced experimental facilities, the same CFD tools can be used in the design of those facilities. This interconnection improves the overall performance of the experimental setup and validation.

The goal of the INFRAL project is to ensure the availability of infrastructures and research teams capable to design, construct and operate test facilities representing the physics of nuclear safety related phenomena with sufficient accuracy. Adopting and testing new, advanced measuring techniques enables to produce high quality test data for the development and validation of modern computational tools.

Adopting and testing the advanced and combined use of new measuring techniques allows targeted research projects to achieve high quality data with using pre-tested configurations of instrumentation and measuring systems. Without this expertise and knowledge such a research may not even be possible or at least not achieving the expected results. Ability to apply the combined use of different advanced systems or using Wire-Mesh Sensor (WMS) in a new application are examples of the project outcomes.

Post-processing of the measurement data is an important part of the research workflow and the development of the data analysis methods is needed if there are no readily available computational tools. Commonly, in-house developed tools are applied to extract the essential information from the WMS and HSC data. In PIV, the postprocessing software is typically supplied to the user by the manufacturer of the PIV system. Along the hardware acquisitions, the development of the in-house data analysis tools has been carried out and it will be continued in the proposed project. The data analysis tools can be quite research problem specific, which means that they have to be customized and further developed for different experimental arrangements. For example, the same numerical procedures applied to analyse the void fraction measurement with WMS cannot be directly applied to extract the data from the flow mixing experiments.

The maintenance work of (PWR) PACTEL consists of the maintenance of the hardware of the facility (piping, vessels, and inspections) as well as the transducers and other instrumentation and the data acquisition system. By this work, the operability of the facilities will be ensured. Besides the (PWR) PACTEL, the laboratory has several control and data acquisition systems, which occasionally require spare parts or even reserve parts to make sure that those parts are available when needed.
The work packages of INFRAL are:

- Advanced measurements techniques (WP1)
- Maintenance and equipment (WP2)
- Modular Integral Test facility MOTEL (WP3)
- Project management and international cooperation and publications (WP4)

In 2017, the spray experiments will be carried out to support the INSTAB project. In addition, Particle Image Velocimetry system (PIV) will be applied for the 2D/3D velocity field measurements in the PPOOLEX experiments when found feasible.

It is foreseen that the results from all activities performed in the project can be widely applied to experimental research performed in LUT and in Finland. The measurement techniques acquired, tested and developed in the project are available for SAFIR and other projects, conducting tests at LUT laboratories. The (PWR) PACTEL facility is in active use in SAFIR projects and also internationally. The INFRAL project ensures that advanced instrumentation and access to integral test facilities is possible also in the future.

Figure 2.4.1.1. Spectrogram of the DCC-05-4 experiment (PPOOLEX). The whole 48-second experiment was divided into 0.5 seconds short time Fourier transforms. The change of the different frequencies during the time are clearly visible.
Jules Horowitz Reactor (JHR), a new European material testing reactor (MTR), is currently under construction at CEA Cadarache research centre in France. Finland is participating in the construction with a 2% in-kind contribution, which includes Underwater Gamma spectrometry and X-ray radiography (UGXR) and Hot-cell Gamma spectrometry and X-ray radiography (HGXR) systems as well as a Mechanical Loading Device for Irradiation Experiments (MeLoDIE). With this in-kind contribution, Finland will have the possibility of utilising the new JHR research infrastructure dedicated to nuclear safety related research. Furthermore, the in-kind contribution enables access to the results of the future experiments.

JHR is designed to provide a high neutron flux (twice as large as the maximum available today in MTRs), to run highly instrumented experiments to support advanced modelling giving prediction beyond experimental points, and to operate experimental devices giving environmental conditions such as pressure, temperature, flux, and coolant chemistry relevant for example for water reactors, for gas cooled thermal or fast reactors, and for sodium fast reactors.

According to the current schedule the construction of the JHR will be ready in 2021 and the first experiments will start in 2022. The planning of these experiments has already been launched within three WGs, namely Fuel WG, Materials WG, and Technology WG, aiming first to a pre-JHR programme carried out in the existing European nuclear facilities, and through this work aiming at clarifying the experimental parameters and conditions needed for the JHR. The objectives of these working groups 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 European research project FIJHOP, gathering a total of 20 partners including the 12 from the JHR consortium, is structured on the two following scientific challenges identified by the JHR Working Groups on Fuel and Material R&D topics:

- FIJHOP-F studying the quantification of phenomena activated in an LWR fuel rod during power transients with focus on those impacting cladding loading and limiting core management in a power reactor;
- FIJHOP-M investigating the neutron spectrum effects on the degradation of low alloy steels (RPV) and stainless steels (reactor internals), and the dose-damage relationship quantified by microstructural characterization and mechanical testing.

Considering the well-integrated MTRs network within Europe, it has been acknowledged by the JHR International Consortium as well as by the NUGENIA community that there is particular importance to prepare, by using today’s operating MTRs, future joint international programs that will be performed in JHR. This is the main general objective of the FIJHOP proposal.

The participation in the three working groups 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 information on the experimental capacity of JHR acquired in the WGs can be used as a guideline in the planning of new experiments. The WGs will assist the JHR research
programme management in planning and realisation of the experimental campaigns. In addition to bringing out our own interests and needs, the WG discussion about general experimental needs and possibilities is useful when making decisions about the participation and collaboration in the future programmes. Furthermore, these needs can be used in the development of new experimental devices in the on-going design phase, which further helps to plan appropriate experiments. The findings and results of WG work will be available immediately, and the information will be specified and expanded as the work progresses.

![Image of Jules Horowitz Reactor](image)

**Figure 2.4.2.1. Jules Horowitz Reactor**

### 2.4.3 RADLAB - Radiological laboratory commissioning 2017

In this third year of the SAFIR2018 research programme, the technical commissioning of the VTT Centre for Nuclear Safety (CNS) will be completed as the hot cells are installed, and the final radiological commissioning will be well underway. The VTT CNS and its hot cell facility is a national infrastructure hosted by VTT, and is considered an important element in fulfilling the national requirements for independent competencies for domestic nuclear power generation.

This RADLAB project was preceded by the REHOT project in the SAFIR 2014 program and in the first year of the SAFIR 2018 program. The RADLAB project is an integral part of the overall infrastructure renewal process surrounding the VTT CNS, in support of both reactor safety and nuclear waste management (NWM) research. While the former REHOT project focused mainly on the design, construction and equipping of the new CNS facilities, this RADLAB project spans the move from the existing facilities at Otakaari 3 (OK3), to the new facilities, and features the commissioning of equipment and ramping up of the infrastructure in the new facilities. In 2017 there is a special emphasis on nuclear waste management research infrastructure commissioning, expanding the high resolution radioanalytical readiness and materials testing capabilities of the CNS, and also renewing and improving the condition monitoring of the experiments in aerobic and anaerobic experiment environments.

The RADLAB project involves efforts in five main areas: 1) hot cell fabrication, installation and commissioning; 2) hot laboratory equipment procurement and nuclearization; 3) design, fabrication and installation of self-built research facilities; 4) design, fabrication and
installation of materials handling and storage facilities; and finally, 5) management of the full laboratory infrastructure commissioning and ramp-up of operations for both reactor safety and final repository research.

Figure 2.4.3.1. Radiological laboratory infrastructure renewal process comprised of simultaneous decommissioning of facilities at Otakaari 3 (OK3) and equipping and commissioning of the Centre for Nuclear Safety.

The infrastructure for radioactive materials research and testing involves facilities, equipment and competent users. The RADLAB project is the means by which the infrastructure investments are executed, supporting the personnel involved in carrying out the work. This includes design input and oversight of ITD in designing and manufacturing the hot cells, but also in training of personnel, adopting a new safety culture, executing the key equipment procurement processes, nuclearization of equipment going into the cells, and the design, procurement and installation of the other research devices and process equipment supporting the radioactive materials handling and storage.

In 2017 the researchers from the various disciplines are sharing the same office building and the new nuclear waste management and radiochemistry laboratory facilities are already functional. Thus, the focus can turn to exploiting the modern equipment and tools to ensure continued high-quality nuclear waste research, while also diversifying the methods for a broader and fuller chain of analytical services.

The ultimate objective of the hot cell contract with ITD is to achieve safe, functional hot cells in a cost effective manner, which are appropriate to the specified research and testing needs. The detailed design was completed on schedule at the end of 2015, and manufacturing of the units has taken place in 2016. In accordance with the current work plan laid out by ITD (Appendix A of this document), installation and technical commissioning of the hot cells at VTT is on track to occur mainly during the first half of 2017.
The primary objective of the equipment procurement is to acquire the most suitable and cost effective hot laboratory, hot cell and ancillary devices and instruments for the specified needs.

The overall objective of the equipment nuclearization and installation (whether purchased or self-built), is to achieve safe functionality of the device in its application for radioactive material handling or testing, whether it is a self-built “hot” autoclave system, a stand-alone device like a “hot” SEM, or a device deployed inside one of the hot cell chambers. In the case of the hot cell suite manufactured by ITD, a full Factory Acceptance Test (FAT) and Site Acceptance Test (SAT) procedure is specified in the contract.

While the principle goal of the RADLAB project is to execute the infrastructure renewal, the exploitation of that result is firstly achieved by demonstration of the functionality of the facilities for producing mechanical and microstructural data and results of radioactive materials in conditions that are in line with the ALARA principle expected of contemporary radiological facilities. The initial exploitation is by the complementary European commission funded research project SOTERIA, the Academy of Finland funded MENUCHAR project and the BREDA project. The goal of the programme period will be to demonstrate the research capacity of the facility, for use in increasing the overall understanding of the effect of radiation on nuclear power plant structural materials.

Figure 2.4.3.2. The RADLAB project produces the infrastructure that is then utilized by research projects associated with the VTT CNS, bringing together experimental and modelling work, and utilizing test-reactor irradiations.
3. Financial and statistical information

The planned total budget of the programme in 2017 is 6.7 M€. The major funding organisations are VYR with 4.05 M€, VTT with 1.42 M€, LUT with 0.18 M€, Aalto with 0.13 M€, Halden Reactor Project with 0.14 M€, NKS with 0.13 M€, and SSM with 0.07 M€ (Figure 3.1). Funding from KYT2018 programme is 0.14 M€ (RADLAB project) and from other organisations 0.47 M€ (including direct funding of some projects by TVO, Fennovoima and Fortum). The volume, funding and costs of SAFIR2018 research projects in 2017 are shown in Table 3.1. The total volume of the programme in 2017 is planned to be 42 person years. The personnel costs make up the major share of the planned expenses (Figure 3.2).

![Planned total funding in 2017 (6,7 M€)](image)

Figure 3.1. Financing of the SAFIR2018 programme in 2017.

Figure 3.3 shows VYR and total funding in the projects of the main research areas of SAFIR2018:

- SG1: Plant safety and systems engineering (steering group)
- SG2: Reactor safety (steering group)
- SG3: Structural safety and materials (steering group)
- RG6: Research infrastructure (reference and steering group).

Figures 3.4 and 3.5 illustrate the distribution of total and VYR funding to the above research areas, respectively.

Figure 3.6 shows the planned person years in 2017 of the projects in SAFIR2018 research areas and Figure 3.7 in reference group areas RG1-RG6:

- RG1: Automation, organisation and human factors
- RG2: Severe accidents and risk analysis
- RG3: Reactor and fuel
The costs related to experimental equipment, materials and external services are reflected in the smaller share of person years versus share of funding in the research infrastructure research area RG6 and SG3 area (Figures 3.4 and 3.6). Membership fees for several international projects are funded in SG2 area (see Table 3.1).

**Figure 3.2.** Cost structure of the SAFIR2018 programme in 2017.

**Figure 3.3.** VYR and total funding in SAFIR2018 research areas in 2017.
Figure 3.4. Total funding in SAFIR2018 research areas in 2017.

Figure 3.5. VYR funding in SAFIR2018 research areas in 2017.
Figure 3.6. Planned person years in SAFIR2018 research areas in 2017.

Figure 3.7. Planned person years in SAFIR2018 reference group areas in 2017.
Table 3.1. Resource plan summary of SAFIR2018 research projects in 2017. Each project belongs to one research area (SG) and one reference group (RG), for details please see SAFIR2018 website (link, SAFIR2018 Reference groups and projects).
4. Organisation and management

SAFIR2018 organisation and is shown in Figure 4.1 and its function described in detail in the Operational management handbook (see SAFIR2018 website: http://safir2018.vtt.fi/).

Figure 4.1. Structure of SAFIR2018 organisation. Each project belongs to one reference group (RG) and its topic may be related to one or several research areas. The reference group RG6 “Research infrastructure” has a special role of a reference and steering group. No additional reference groups (RG7-RGN) have been planned in 2017.

The programme management bodies, the management board (MB), the steering groups (SG) and the reference groups (RG), have meetings on a regular basis. The steering and reference groups have at least three meetings annually for following the progress of the projects. The management board also has at least three meetings during the year. Project meetings are encouraged for discussing specific topics of the projects. The programme staff and their main duties are listed in Appendix 1. The persons involved in the management board, the steering and reference groups are listed in Appendix 2.

The SAFIR2018 management board can annually initiate small preliminary type studies with the order procedure. Decisions on the small projects 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 and they can also introduce new topics. In 2017 two projects have been planned. The costs of small projects are included in the budget of the administration project (ADMIRE) as subcontracting. The final reports of previous small projects can be found on SAFIR2018 extranet.

The information on the research performed in SAFIR2018 is communicated formally via the progress reports of the projects for the reference group meetings, the reports of the programme [3,4,5], and SAFIR2018 website (public and protected extranet). Additional information is given in seminars organised in the various research areas. The detailed scientific results are published as articles in scientific journals, conference papers, and research reports.
References


Appendix 1

SAFIR2018 project and resource plans 2017
Steering Group SG1 -

Plant safety and systems engineering:
SAFIR2018 Project plan
Version 1.0

CORE
Crafting Operational RESilience in nuclear domain

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1. Research theme and motivation

1.1 Background and state-of-the-art

Both nationally and internationally, there is a period of moratorium going on in the nuclear industry. Many new-build projects have been postponed, and there have been changes in upgrade projects and delays in digitalization of automation systems and control rooms. In part, these delays and changes are caused by the fact that safety requirements have become tighter in the aftermath of the Fukushima accident. Nuclear companies have to manage the increased complexity, technological, environmental and social changes, and at the same time meet the increased safety requirements. In addition, there are also political and economic challenges that they have to face and that have had an impact on this moratorium stalemate.

The present situation has implications for research needs in the area of Human Factors (HF). The research should focus to a larger extent on safety and performance of running nuclear power plants (NPPs), and specifically on the improvement of operational capabilities and practices of operating personnel in different plant states. According to the international evaluation of SAFIR2014 programme, the work on control rooms and operator practices should be more focused on operator and safety requirements rather than on control-room technologies, and there should be a closer collaboration between research activities in areas of control-room work and man, organization and society. According to the SAFIR2018 framework plan (MEE, 2014), since the operating personnel are required to have very in-depth understanding of plant systems and their operation, there is a need for research concerning competence development in highly regulated and automated work. It is also stated that solutions have to be developed that help the operating personnel to anticipate upcoming situations and plan how to cope with their consequences. In addition, research on the area of Human Factors must take into account the defence-in-depth principle and the development of overall safety.

According to the update to the Framework plan for 2017, the themes described in the memorandum of an ad hoc planning group (“SAFIR2018 ja ajankohtaiset Tepco Fukushima Dai-ichi onnettomuudesta johdetut kansainväliset tutkimusteemat”) should be taken into account in preparing the proposals. In addition, there will be a change in the electricity production modes supporting the grid: Less conventional base load capacity will be available for maintaining the power balance, and this may also have effects on the operation of the nuclear power plants. Therefore, potential safety issues related to the operation of nuclear power plants in the load following mode are an important topic for research. For example, possible operation of nuclear power plants in the load following mode could have effect on operator work and training and safety I&C. According to the update, the operation of the emergency preparedness organisation and co-operation and interactions between different organisations should also be emphasised in the research.

In previous SAFIR programmes, Human Factors research has been divided into two research areas: One part of the research has been mainly focused on organisational issues, and another on human-system interaction and work research. In the future, our aim is to channel Human Factors research efforts to topics which break new grounds in SAFIR-funded research and which are better anchored to the development of nuclear safety, but at the same time are founded on research that has been conducted in previous Human Factors -related SAFIR-projects.

The defence-in-depth concept provides a general framework for analysing of research needs in the nuclear domain. The approach includes the prevention of abnormal conditions and their degradation and mitigation of their harmful consequences. Typically, five depth levels have been identified (e.g., Liebman, 1996), but from the Human Factors point of view it is sufficient to consider the following three levels: prevention, preparation, and consequence management. ‘Prevention’ refers to systematic assessment of risks and design of prevention mechanisms for known risks; ‘preparation’ means the establishment of structures and resources for inevitable risks; and in ‘consequence management’ it is critical to minimize the damage and prevent critical systems from breaking down (Boin et al., 2010). In order to develop good practices for the promotion of nuclear safety, research has to focus on each of these three stages in accident prevention and management.

There are contrasting views as to what the best ways are to promote nuclear safety. According to the traditional approach to safety, it can be defined as a condition where the probability that things go wrong is very low and the
aim is to prevent that something goes wrong (Hollnagel, 2006; 2011). It is also proposed that the best way to do that is to build barriers to prevent fatal errors and to establish detailed guidance for operating personnel. However, according to many authors and practitioners, this is not enough, because by in this way the operating personnel are not able to cope with the complexity of socio-technical systems and to prevent accidents effectively (e.g., Hollnagel, 2011a; Woods, 2006). A nuclear power process is a complex socio-technical system consisting of large number of human and technical elements, surrounded by a natural environment and society. It is characteristic for this kind of complex systems that there are strong couplings between elements, so that a failure in one element may have cascading influences throughout the system. There are also non-linear relationships between elements which mean that a small disturbance may cause a large effect and emergent behaviours that cannot be fully anticipated.

In order to cope with this kind of interactional complexity, we need a systemic view on safety and complexity. According to this view, both normal performance and failures are emergent phenomena, so they cannot be understood by referring to errors caused by individual operators (e.g. Hollnagel, 2006; 2011a). Therefore, regarding human behaviour, we should not study errors as a separate topic but in the context of normal human behaviour in real work situations and the mechanisms that are involved in adaptation and learning (Rasmussen et al., 1987). The adaptability and flexibility of human behaviour is the reason for its successes, but sometimes they also lead to failures (e.g. Hollnagel, 2006; 2011a). According to this view, possible failures are caused by normal variability of context and conditions rather than by failures of actions. This kind of systemic model to safety is based on a functional conception in which safety is conceptualized as the ability of a system or an organization to react to disturbances and recover from them promptly with no impact on the dynamic stability of the system (Wilson, 2014).

The systemic view has several implications for both research and practices in the nuclear domain (Dekker, 2011; Hollnagel, 2006; 2011a; Woods, 2006): first, it is necessary to look at successes as well as at failures in order to understand failures and why things go wrong; second, it is necessary to look at weak signals, i.e. early signals of problems that start to occur and identify indicators and markers that are considered as important to safety; thirdly, it is necessary to anticipate future events and conditions that may have an impact on the system’s ability to function adequately; and fourth, it is also necessary to distinguish the need to learn new ways to adapt and adjust adaptive capacities. Overall, there is a need for novel models and conceptualizations of operator activity that enable safety and reliability in different plant states, from normal operation through anticipated design basis events to unanticipated beyond design basis events.

1.1.1 Our expertise in Human Factors of nuclear safety

VTT has over 30 years of experience of Human Factors research in the nuclear domain. The project team’s approach is based on interplay of three kinds of research, design, and innovation activities: empirical research to understand current situation in the system; analytical modelling to model and simulate alternative development paths; and participatory development accomplished through interventions in which the actors gain understanding of the potential future and are then empowered to make decisions to carry out transformations. The team has developed a systems usability framework for evaluating tools in control room work (e.g., Savioja & Norros, 2013; Savioja, 2014), and recently the team has, for example, studied operators’ conceptions of procedure guidance in NPP process control (Norros et al., 2014), and the effect of user interface solutions on the development of automation awareness (Karvonen et al., 2014). The team has also developed an integrated approach to control room lifecycle management, including tools and methods to tackle challenges related to different stages of the design process (Laarni et al., 2014). In addition, VTT has experience in studying operational and organizational resilience (e.g., Macchi et al., 2011; Savioja et al., 2014).

Finnish Institute of Occupational Health (FIOH) is the leading occupational health research centre in Finland and its goal is to study and improve the health and wellbeing of workers. In the research area of Brain at Work, large data sets on health, brain activity, and cognitive capabilities have lately been collected. The Brain and Technology Research at FIOH has over 10 years’ experience of carrying out experimental laboratory studies in which neurophysiological metrics, work simulation, computerized work tasks, cognitive tests, subjective symptoms questionnaires, and immunological, metabolic and hormonal tests have been integrated in varying setups depending on the research questions and/or methodological development challenges of individual projects.

FIOH also has a long history in work place developmental interventions and in modelling work processes and activities for improving work-related wellbeing (Leppänen, 2001; Leppänen et al., 2008; Seppänen et al., 2009). The WP2 project team at FIÖH has a long and recognized experience in the design and realization of participatory developmental interventions (Schaupp et al., 2013). Moreover, it has expertise in the investigation of workplace learning (Seppänen, 2002; Seppänen, 2004; Pereira-Querol et al., 2010) as well as in the examination and creation of methods for developing capabilities (Schaupp, 2011; Seppänen & Kloetzer, 2014; Virkkunen & Schaupp, 2011). These, together with the experience in the analysis of safety critical work activities (Seppänen et al., 2014), will significantly advance the aims of CORE, in particular in WP2.
The project work in most of the work packages adheres to a certain order that is based on how research and development work is structured in action-centred approaches (e.g., Change Laboratory, Virkkunen & Newham, 2013). The aim of the project (see Figure 1) is to

1) chart the situations, i.e., gather operational experiences and conduct observations and interviews in the beginning of the project
2) analyse the collected data, and model and simulate human cognition/behaviour, distributed cognition and communication and collaboration
3) develop interventions and training activities

1.1.2 General objectives of the project

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 self-reflexive, engaged, and self-conscious and aware of high-level goals, instead of being solely guided by fixed and predetermined procedures. Skills and competencies are needed to better manage task switching and distractions/interruptions in dynamic multitasking operational environments, and troubleshooting should be a more collaborative enterprise. The aim is also to develop new Human Factors guidelines, models and tools and preventive 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.
4) examine and test the developed interventions and
5) develop guidance and consolidate proposed practices.

Figure 1. The continuum of research activities in most of the WPs in the CORE project.

The project is divided into six work packages with the aim to consider operational resilience from different perspectives. The project activities throughout the SAFIR2018 programme period are depicted in Figure 2. In WP1, tools and practices are developed for gathering positive operating experiences from challenging operational situations, for systematization and review of these experiences and for refining the lessons learned for training purposes. In WP2, the aim is to promote proactive and prudent attitude among operational personnel both at the individual and team level by supporting reflective thinking and learning. In WP3, the aim is to promote operational resilience in multitasking, diagnostic reasoning and decision making situations by developing tools and practices for effective interruption management, troubleshooting and self-reflective learning. In WP4, interventions and guidance are developed for the management of acute stress and fatigue; furthermore, methods are refined for the online measurement of workload and fatigue in simulation tests. In WP5, the aim is to develop guidance especially regarding the communication and coordination of activities in emergency exercises, with the final objective of enabling the licensees and stakeholders to better meet the demands which presently known and also unknown threats pose to the nuclear power plants. In WP6, the aim is to broaden the view of safety competence of the operators and experts of the nuclear domain with the human contribution point of view. Current Human Factors guidelines and practices are evaluated, and new HF tools will be tested and examined in interventions, workshops and every day safety management work.
The project plans have been discussed with Human Factors experts at TVO, Fortum and STUK, and some of the research topics have been suggested by power companies. The research work will be conducted in co-operation between VTT and FIOH.

The six work packages are linked to at least one defence level, and most of them can be linked to several ones as shown in Table 1.

---

**Figure 2. Project activities throughout the SAFIR2018 programme period.**

<table>
<thead>
<tr>
<th>TASK</th>
<th>2015</th>
<th>2016</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
<tbody>
<tr>
<td>WP1: LEARNING FROM SUCCESSES</td>
<td>Data collection First principles determination</td>
<td>Method &amp; guidance development</td>
<td>Consolidation of lessons learned</td>
<td>Further development of CER system</td>
</tr>
<tr>
<td>WP2: WORK-BASED LEARNING</td>
<td>Baseline data collection &amp; exploration</td>
<td>First development method try-out</td>
<td>Evaluation of method’s impact</td>
<td>Subjective evaluation &amp; productization</td>
</tr>
<tr>
<td>WP3: SUPPORTING DISTRIBUTED COGNITION</td>
<td>Review on multitasking &amp; diagnostic reasoning</td>
<td>Modelling distributed cognition</td>
<td>Impact of goal conflicts and heuristics</td>
<td>Guidance development</td>
</tr>
<tr>
<td>WP4: SUPPORTING STRESS MANAGEMENT</td>
<td>Baseline data collection</td>
<td>Data analysis Stress management</td>
<td>Effects of stress management Modelling instructor training</td>
<td>Guidance development Stress in outages</td>
</tr>
<tr>
<td>WP5: SEVERE ACCIDENT MANAGEMENT</td>
<td>State-of-the-art of emergency exercises</td>
<td>Interviews Modelling</td>
<td>Data collection Intervention development</td>
<td>Guidance development</td>
</tr>
<tr>
<td>WP6: HUMTOOL</td>
<td>Modifying &amp; testing HF-tool Id of current practices</td>
<td>HF-tool validation HF need recognition</td>
<td>HF-tool interventions &amp; development</td>
<td>Evaluation of HF-tool impl &amp; future needs</td>
</tr>
</tbody>
</table>
Table 1. WPs in terms of levels of defence.

<table>
<thead>
<tr>
<th>WP</th>
<th>PREVENTION</th>
<th>PREPARATION</th>
<th>CONSEQUENCE MANAGEMENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>WP1: Operating experiences</td>
<td>Resilience indicators</td>
<td>Resilience indicators</td>
<td></td>
</tr>
<tr>
<td>WP2: Work-based learning</td>
<td>Reflective thinking</td>
<td>Reflective thinking</td>
<td></td>
</tr>
<tr>
<td>WP3: Supporting distributed cognition</td>
<td>Fluent multitasking</td>
<td>Fluent multitasking Efficient troubleshooting</td>
<td>Fluent multitasking</td>
</tr>
<tr>
<td>WP4: Supporting stress management</td>
<td>Efficient coping with stress</td>
<td>Efficient coping with stress</td>
<td></td>
</tr>
<tr>
<td>WP5: Supporting severe accident management</td>
<td>Identification of contributing HF</td>
<td>Self reflection</td>
<td>Resilient co-operation</td>
</tr>
<tr>
<td>WP6: HUMTOOL</td>
<td>Identification of contributing HF</td>
<td>Self reflection</td>
<td>Skills to analyse and learn contributing HF</td>
</tr>
</tbody>
</table>

1.2 Objectives and expected results

In general, the aim of the project is to improve safe operation of nuclear power plants by developing guidance, good operational practices and other practical solutions that promote resilience in nuclear operations. In order to reach this objective, we will develop new guidelines, models and simulations, and training interventions that will be tested and examined in simulated test environments. More detailed objectives and results are presented below.

The main objectives of the project are the following:
- understand how to capture successful performance adaptations from operating events and how to analyse and communicate the lessons learned from successes
- develop new concepts to describe operator work and develop means for promoting reflectivity and work-based learning
- investigate the role of multitasking and effects of interruptions in operative work
- analyse the reasoning and problem solving processes in difficult troubleshooting situations
- promote self-reflection in training by self-assessment
- investigate the effects of acute stress on operator performance in simulated accident situations
- understand how emergency exercises are planned and conducted and investigate the resilience markers and developmental needs, especially the ones related to communication and coordination, in emergency exercises
- study appropriateness and use of the HF tool as an investigation method in operative event analysis
- evaluate the implementation process of the HF tool, its effects on safety thinking, as well as supporting and hindering factors of the process.

The main results are the following:
- method for analysing successful aspects of human and organisational performance and guidance on how to share the lessons learned from successful performance adaptations with other NPPs
- new training methods for promoting resilience in operators work practices
- better understanding of how to support multitasking and interruption management in operator work
- better understanding of operator/maintenance personnel’s thought processes in diagnostic reasoning
- training material and/or programme for complex troubleshooting
- better understanding of operator stress and its management in incident and accident situations and new methods for supporting operators to manage their stress in demanding situations
- guidance for developing the processes used in emergency exercises and especially for building efficient collaboration and cooperation in emergency exercises
- new HF tool to be used in identifying human contribution in operative events
- recognizing the human contribution in current nuclear safety management system.
1.3 Synergies between work packages

The six work packages of the CORE project converge to a common target, and therefore, there are synergies between the work packages. The WP1, 2, 5 and 6 have common aims and ideas concerning learning and analysis of safety capabilities (see Figure 3); respectively, WP3 and 4 have a common aim to promote operational resilience by developing tools that foster effective task sharing, tactful communication, fluent troubleshooting, and optimization of cognitive load and level of stress (see Figure 4).

The six WPs address the topic of operational resilience from different perspectives in order to foster human capabilities in the nuclear domain, i.e., learning, management and analysis of human factors of own actions. These perspectives are targeting each of the three levels of defence-in-depth, i.e., prevention, management, mitigation as well as aftermath of abnormal conditions (see Table 1 above).

---

**Figure 3.** The process of utilizing WP1, WP2, WP5 and WP 6 actions and results in crafting operational resilience of nuclear operators.

**Figure 4.** The process of utilizing WP3 and WP4 actions and results in crafting operational resilience of nuclear operators.
A joint series of workshops will be arranged between different WPs during 2017 to exchange experiences between the researchers. In those workshops, research findings of each WP will be shared, joint scientific and practical outputs will be planned, and efforts will be made to outline a commonly shared ‘big picture’ of the current state and future needs regarding the operational resilience in nuclear domain, in order to:

- find common issues and threads concerning the operational resilience, and to be further communicated as a part of CORE reporting in RG meetings and scientific outputs of the CORE project;
- promote new understanding and ways of thinking in the nuclear domain, both for regulator and plant management and operations.

The synergy value based on similarities and differences between the WPs is depicted in Table 2.

**Table 2. Similarities, differences and synergy between the CORE work packages.**

<table>
<thead>
<tr>
<th>WP</th>
<th>Similarities and overlapping themes</th>
<th>Differences</th>
<th>Synergy (possible joint actions)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>1 Operating experiences</strong></td>
<td>Providing a methodology to support OE activities (WP3)</td>
<td>Focus specifically on defining and capturing successful adaptations</td>
<td>Complementing event analysis method proposed in WP3 with principles for capturing successful adaptations</td>
</tr>
<tr>
<td></td>
<td>Continuous learning and the collection of experience data (WP2, WP6)</td>
<td>Organisational learning and knowledge transfer of successes from OE perspective</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Exploring the possibility of promoting learning from successes by means of positive psychology</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>2 Work-based learning</strong></td>
<td>Support to distributed cognition (WP3)</td>
<td>Interpretative work (WP2)</td>
<td>Cognitive modelling with WP3 focused on cognitive processes with WP3 &amp; 4</td>
</tr>
<tr>
<td></td>
<td>Need to analyse work processes (WP2, 5)</td>
<td>Need to analyse work processes (WP2, 5)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Modelling communication and collaboration (WP5)</td>
<td>Modelling communication and collaboration (WP5)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Acute stress (WP4)</td>
<td>Acute stress (WP4)</td>
<td></td>
</tr>
<tr>
<td><strong>3 Supporting distributed cognition</strong></td>
<td>Interpretative work (WP2)</td>
<td>Cognitive modeling</td>
<td>Collecting data with WP4</td>
</tr>
<tr>
<td></td>
<td>Need to analyse work processes (WP2, 5)</td>
<td>Focus on cognitive processes</td>
<td>Joint workshops with WP2 &amp; 4</td>
</tr>
<tr>
<td></td>
<td>Modelling communication and collaboration (WP5)</td>
<td></td>
<td>Reporting results with WP2 &amp; 4</td>
</tr>
<tr>
<td></td>
<td>Acute stress (WP4)</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>4 Supporting stress mgmt</strong></td>
<td>Need to model the operator work and evaluate performance (WP3)</td>
<td>Quantitative physiological recordings of acute stress</td>
<td>Collecting data with WP3</td>
</tr>
<tr>
<td></td>
<td>Need to model perceived stress and lead (WP2)</td>
<td>Stress management based on physiology</td>
<td>Analysing data: joint evaluation of operator performance and stress with WP3</td>
</tr>
<tr>
<td></td>
<td>Effect of acute stress on performance (WP3)</td>
<td>Evaluation of management and training effectiveness with quantitative physiological recordings</td>
<td>Joint workshops with WP2 &amp; 3</td>
</tr>
<tr>
<td><strong>5 Supporting severe accident mgmt</strong></td>
<td>Need to analyse work processes (WP1, 2, 5, 6)</td>
<td>Focus on collaborative activities among various stakeholders, not only among plant personnel</td>
<td>Reporting results with WP3</td>
</tr>
<tr>
<td></td>
<td>Modelling communication and collaboration (WP3)</td>
<td></td>
<td>Reporting results with WP3 (communication and collaboration).</td>
</tr>
<tr>
<td><strong>6 HUMTOOL</strong></td>
<td>Providing a methodology to support OE activities (WP1)</td>
<td>Proving a (ready) tool for nuclear domain to understand the overall field of ifps in safety critical organisation, from individual to organisational level</td>
<td>Collecting data</td>
</tr>
<tr>
<td></td>
<td>Supporting continuous learning and the collection of experience data (WP1, WP2)</td>
<td>Supporting a capability of the organisation to collect and summarise their safety data for further actions</td>
<td>Joint workshops, e.g., analysing successes (workshops and training), supporting skills of self-reflection (workshops and interventions)</td>
</tr>
<tr>
<td></td>
<td>Focusing also on successes (WP1)</td>
<td></td>
<td>Reporting results with other WPs, e.g., with WP1</td>
</tr>
<tr>
<td></td>
<td>Need to analyse work processes (WP2, 5)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
1.4 National and international co-operation

According to the update to the Framework plan for 2017, research organisations are encouraged to seek for still more co-operation with other national and international research organisations in relevant topics that are of common interest.

The research in this project is associated with several activities central to the Human Factors Engineering program such as operating experience review, staffing and qualifications, treatment of important human actions, procedure and training development and verification and validation (cf., NUREG-0711; YVL B.1). The project has links to other projects in plant safety area, i.e., SAUNA, PRAMEA and MAPS. Especially, the SAUNA project includes a task devoted to Human Factors Safety Case development, and the PRAMEA project includes a task devoted to HRA method development. These tasks will profit from the results of the CORE project, and some work done in the SAUNA project may have an impact on the work in CORE. There will also be a deepening collaboration between the CORE and MAPS project: Task T3.3 of MAPS (“Safety culture development methods in subcontractor networks”) and WP1 of CORE both focus on practical organizational methods for improving safety. There will be collaboration between the tasks by means of information exchange to ensure synergies. In addition, the MAPS project and WP6 of the CORE project will organize a joint workshop in 2017.

Collaboration with Halden Reactor Project (HRP)/Institute for Energy Technology (IFE) will continue as has been done before. The collaboration is in part accomplished through NKS projects. NKS funding has been applied for WP1 and WP3 in 2017. In addition, Halden In-kind funding is allocated to emergency exercise research (WP5) and collaboration between Halden and VTT in this work package is established (described in the work plan as part of WP5 description). Also in WP4, there will be collaboration with HRP/IFE in method development, data collection and analysis in 2017.

The project aims to collaborate with EdF (Electricité de France), the Human Factors research team at IRSN (L’Institut de Radioprotection et de Sûreté Nucléaire) and with the stakeholders of the NUGENIA (Nuclear Generation II & III Association). In addition, it has been planned to continue collaboration with the following universities, and research institutes: Chalmers University, Universite Lumiere 2 (Lyon), University of Neuchâtel and University of Oxford. The project work will also contribute to OECD/NEA Working Group of Human and Organisational Factors (WGHOF). At present, Jari Laarmi (VTT) is the Finnish delegate in the OECD/NEA/WGHOF and NUGENIA (HOF area).

The Swedish nuclear authority (Strålsäkerhetsmyndigheten, SSM) has participated in SAFIR reference group work in 2010-2014, and it will participate in SAFIR work also in the future. Collaboration with SSM in the Human Factors area has been planned in private meetings, and new discussions will be arranged in 2017.

1.5 Exploitation of the results

The project is targeted to develop operational practices that promote resilience in different levels of defence and that can be utilized by design organizations, nuclear power plant utilities and regulatory bodies. The project will develop guidelines for prevention, preparation and consequence management at the operational level and training interventions and tools that complement the existing guidance and training procedures. The project provides solutions and practices for better management of severe unanticipated accidents and challenging incident situations. These results will further improve nuclear safety at the national level by improving ability to respond, monitor and anticipate to potentially disruptive situations at the operational level.

The objective of all work packages is the immediate exploitation of project results. It is also aimed to develop new work practices to promote collaboration and communication between researchers and different stakeholders (e.g., representatives of design and operational units of power companies and of the Finnish nuclear authority).

In WP1, a guideline to capture and promote lessons learned from successful performance are developed and piloted. In WP2, an self-evaluation method promoting interpretative orientation to operator work is developed, and it will be deployed for training assessment in NPPs. In WP3, new methods, tools and techniques are designed that will be used in development of human relations practices, in training of operative personnel and in the design and evaluation of licence examination. In WP4, training material and courses on stress management in operative work are developed and delivered/launched. In WP5, regarding emergency exercises and, accordingly, the readiness to meet threats, the project will deliver information to the licensees to improve their practices during an emergency.

In WP6, a new practical human factors investigation tool and training are introduced to nuclear domain, aiming to broaden the HF thinking and to improve the comprehension of human contribution as a part of safety management.
1.6 Appropriateness of the project to SAFIR2018 programme

The research needs of this research area are described in chapter 3.2 of the SAFIR2018 framework plan “Plant safety and systems engineering” and especially in section 3.2.4.2 “Organisation, human and interest groups” (see Table 4 on page 15). The results of the project provide answers to the research needs and challenges that are described in this section. The project is also connected to other subtopics of the plant safety research area, i.e., 3.2.4.1 “Overall safety understanding”, 3.2.4.3 “Operating processes as a support for plant safety” and 3.2.4.4 “Factors affecting technical safety solutions”.

A special attention will be placed on some of the research needs that have been addressed in the update to the Framework plan for 2017 (see Table 3).

Table 3. Treatment of the research needs addressed in the update to the Framework plan.

<table>
<thead>
<tr>
<th>Research theme</th>
<th>Work package</th>
<th>Treatment of the theme</th>
</tr>
</thead>
<tbody>
<tr>
<td>Implications of Tepco Fukushima Dai-ichi accident</td>
<td>WP1</td>
<td>Guideline for capturing lessons learned from successful performance will be applied in document analysis of public event reports of the Fukushima accident</td>
</tr>
<tr>
<td></td>
<td>WP4</td>
<td>Educating operators on the stress-related factors in the Fukushima accident</td>
</tr>
<tr>
<td></td>
<td>WP6</td>
<td>Extending the comprehension of HF in nuclear power industry from the macro-ergonomics and system thinking point of view, to improve the competence of the actors to identify and manage HF related risks on the individual, work, group, organizational and system levels</td>
</tr>
<tr>
<td>Operation of the plant in the load following mode</td>
<td>WP2</td>
<td>Identification of training challenges in the operation of the plant in the load following mode</td>
</tr>
<tr>
<td></td>
<td>WP3</td>
<td>Identification of multitasking challenges and goal conflicts associated with the operation of the plant in the load following mode</td>
</tr>
<tr>
<td>Increasing international cooperation</td>
<td>All WPs</td>
<td>Co-operation with HRP/IFE, Nordic NPPs and SSM</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Participation in OECD-NEA-WGHOF task groups</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Membership in the NUGENIA community</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Participation in H2020 Euratom projects</td>
</tr>
<tr>
<td>Operation of the emergency preparedness organisation</td>
<td>WP6</td>
<td>Delivery of information to the licensees to improve their practices during an emergency</td>
</tr>
<tr>
<td>Safety implications of potential new unknown threats</td>
<td>WP2</td>
<td>Training and preparation to confront unknown threats</td>
</tr>
</tbody>
</table>

1.7 Education of experts

A special attention is paid to the education of new Human Factors experts. At the moment several doctoral students are working in the project and are preparing their doctoral thesis partly on the topics of the project. Three dissertations are estimated to be completed in 2016-2018. Under the scientific spearhead project titled “Modelling and simulation aided licensing of digital nuclear I&C” (MOSILDIC) three doctoral students at VTT will develop methods and tools that combine understanding from different HFE topics.

We will also employ students and provide them opportunities to do their Master’s work in the project. In addition, we will provide opportunities for researchers at VTT and FIOH who have not been involved in SAFIR projects before to participate in research on the nuclear domain.

The WP6 aims to build the proposed 4-year project in such a way that there is a clearly defined slot for one doctoral dissertation from Tampere University of Technology (TUT). The topic for this has been formulated and a more accurate content and structure of the dissertation will be defined at the end of 2015. Furthermore, there is an
option to integrate project findings to university education. We have negotiated a possibility to utilize project results in Lappeenranta University of Technology (LUT).

The previous SAFEX projects (2010-2014, 2006-2009) have already improved the expertise of the personnel that participates to this project (Pahkin et al. 2014; Pahkin et al., 2010). The project manager of the WP6 has previously acted as a chair of the safety committee of Fennovoima (2009-2012) and as a member of SAFIR2014 ‘human, organization and society’ - reference group in 2010-2015, now with the possibility to deepen her understanding of the area through post-doctoral studies and academic outputs.

To summarize, the following education results are reached in WP6:

- processing one doctoral thesis, with the preliminary stated topic ‘Safety learning from incidents in nuclear industry: implementing a new tool in safety critical organisations’. (Puro, Vuokko)
- two international, scientific reviewed articles, regarding the results of the WP6
- contribution of WP6 results in LUT courses.

Table 4. Addressing the research themes and goals of SAFIR2018 (MEE, 2014) in the CORE project.

<table>
<thead>
<tr>
<th>SAFIR2018 Research Themes, Topics and Goals</th>
<th>CORE Research Themes, Topics and Goals</th>
</tr>
</thead>
<tbody>
<tr>
<td>“[…] it is particularly important to understand the mutual interactions of the systems and the characteristics of the resulting whole, i.e., to understand the plant as a ‘system of systems’, which also includes their socio-technical systems. Since human actions are an important part of maintaining safety functions, particularly over a long period of time it must be possible to take into account the risks related to the activity of humans.”</td>
<td>All WP5s of the CORE project consider the risks associated with human behaviour, and all of them endeavor to develop novel methods and tools to tackle these risks and increase nuclear safety. For example, tools are developed to better identify human contribution in operative work (WP5), to foster complex troubleshooting (WP3) and stress management (WP4) and to build a more comprehensive operational picture in emergency situations (WP5).</td>
</tr>
<tr>
<td>“The operating personnel of nuclear power plants are required to have very in-depth understanding of plant systems and their operation. Developing work policies and competence in a highly regulated and automated work is a particular challenge. Research is needed for developing systemic models for assessing the interactions of different social, psychological, organisational, economic and technological factors in the field of nuclear power.”</td>
<td>Work in the CORE project is based on a systemic approach to nuclear safety (see 1.1), and all WP5s try to better understand the complexity of the nuclear domain and develop more ‘contextualized’ models and tools for their target areas.</td>
</tr>
<tr>
<td>“From the perspective of defence in depth, it is important to study the operation of the plant and the joint activity of organisations if the plant must be restored to a safe state under extreme circumstances during or after a storm or earthquake. Fires must also be taken into account in the safety design of a nuclear power plant.”</td>
<td>All WP5s aim to develop new work policies and practices and foster competence in operative work. For example, in WP1 and 2, new practices are developed for reviewing operating experiences and foster work-based learning.</td>
</tr>
<tr>
<td>“Maintaining and developing safety and operational readiness require online monitoring of the activity and safety margins of the plant systems. The monitoring and control of the nuclear power process is done with the user interfaces of the control rooms. The aim must be to develop solutions in control rooms and user interfaces that help the personnel predict upcoming situations and plan how to cope with undesired consequences. With respect to the actions of the operating personnel, it is important to study what kinds of changes new technology (e.g., digital instructions, remote monitoring and mobile solutions) causes in the work practice. Research on control rooms and user interfaces must closely be related to defence in depth and the development of overall safety.”</td>
<td>WP5s 3-5 will investigate the operation of the plant and the joint activity of organisations under severe accidents, and guidance and tools are developed to “craft operational resilience under extreme circumstances.”</td>
</tr>
<tr>
<td>Even though there is no single WP in the CORE project dedicated to study the user interfaces of control rooms, the results of the work done in WP5 and 4 can be applied to design control room systems that help personnel to better anticipate upcoming events and cope with their possible consequences.</td>
<td></td>
</tr>
</tbody>
</table>
2. Work plan

The project is divided into six parallel work packages, and each of them is further divided into several tasks. The work packages are associated with different levels of defence-in-depth as depicted in Table 1. The work packages are the following:

1) Learning from successes in nuclear power plant operation to enhance organizational resilience
2) Developing work-based learning in the NPP domain
3) Supporting operational resilience in complex and dynamic environments
4) Supporting operator performance in extreme stress
5) Supporting resilience in emergency management
6) Applying a HF tool to learn to analyse human contribution to nuclear safety.

2.1 Work package 1 (WP1) Learning from successes in nuclear power plant operation to enhance organisational resilience

One of the cornerstones for achieving safe and efficient nuclear power plant operation is to learn from experiences; experiences gained within the particular plant or in other plants worldwide. Lessons learned are documented as operating experiences (see, e.g., DOE, 2009) and shared between plants by e.g. IAEA, INPO, WANO, and Owner’s Groups. Usually operating experiences will convey lessons learned from unwanted events in order to prevent similar events – incidents or accidents - from happening in the future. This approach also generally reflects how lessons are learned within the nuclear industry (Pietikäinen et al., 2010).

However, this way of learning from experience does not necessarily capture all lessons learned or even the most important lessons learned on how to enhance safety from the broad set of experiences gained by NPPs. Especially, when analysing more complex events, i.e. events that have multiple contributing causes, including human and organisational issues which interact in unexpected ways to produce unforeseen outcomes, a different approach is needed. The reason is that complex events cannot meaningfully be described using simple cause-event chains. Since complex events also tend to be unique, the impact of preventing specific causes and blocking specific links between cause and events may be limited: a similar type of event with slightly different causes and interactions may still happen. In practice, this challenge can be seen in nuclear organisations, which report repeating events with nearly similar human, organisational and cultural contributing factors but for which they seem to lack efficient corrective actions.

This work package aims at improving nuclear safety by enhancing organisational learning from successful actions and decisions. We want to develop an operating experience review method for capturing, analysing and communicating lessons learned based on successes. This implies that the data basis for generating lessons learned in NPPs will be significantly expanded: rather than learning from the 1 out of 10,000 events, which according to generic estimations go wrong in organisations that emphasise performance (Amalberti, 2006), it will be possible to also learn from the 9.999 events that go right (Hollnagel, 2013).

The potential added value of learning from successes will, however, not necessarily be identical for all successful performance outcomes. In relation to simple routine tasks, where performance succeeds, because it is prescribed by comprehensive and complete procedures, the learning potential might be limited. The reason is that these situations have already been analysed, understood, and adequately addressed from a safety perspective. Still, in relation to more complex tasks and/or in situations where unexpected events occur, the learning potential from successes might be substantial: by analysing how such events were handled successfully, i.e., how performance was adapted to achieve the desired outcome, it will be possible to determine and enhance the human and organisational issues that contribute to ensure successes.

In NPP settings, operation is heavily proceduralised. For this reason, performance adaptation typically implies adaptation of procedure use to situational characteristics. Adaptation of procedure use usually involve a set of the informal practices carried out by personnel – prior to, in between and/or post – the procedure steps. It may include, e.g., knowing what type of information, tools and material is likely needed prior to starting a job and ensure that this is available. One lesson learned in the NKS-project MOREMO, which studied maintenance work practices during the outages, was that these types of activities may be perceived as so mundane that people do not pay attention to them: Workers feel that the work proceeded uneventfully even though multiple small proactive adjustments took
place (Oedewald et al., 2012, Gotcheva et al., 2013). A similar lesson was learned in the NKS-project HUMAX: In many cases maintenance personnel found it pointless to carry out a Post-Job Debriefing following a successful task performance process. A typical comment would be: “If a task is solved as planned, there really isn’t much to talk about” (Skjerve & Axelsson, 2014).

The potential safety gains from learning based on successful performance in Nordic NPPs could be significant: It may contribute to expanding and retaining operational staff’s competence by supporting the sharing of knowledge and insights about how to make things work. It may promote staff members’ sensitivity to safety threats, and specifically their ability to anticipate and proactively deal with risks. Based on positive psychology, it can also be expected to spur a still more positive attitude and engagement in safety work: It will increase focus on sharing success stories, rather than a one-sided sharing of information about situations in which the individual and/or others have failed. Finally, it may reduce some of the negative impacts of turn-over: When a staff member leaves the organisation, important aspects of his or her competence, which are lost today, will naturally remain in the organisation as shared know-how.

2.1.1 Specific aims of WP1

The overall aim of the WP1 is to improve nuclear safety by enhancing organisational learning from successful actions and decisions. Specifically, the objective of the WP1 is to develop a method for capturing, analysing and communicating operating experiences based on successful performance adaptation in Nordic NPPs. The approach may broaden current practices for handling operating experiences (e.g. event investigation practices) to enhance safety.

2.1.2 Specific research questions of WP1

- How to learn from successes? What kinds of theoretical models and practical concepts can be found from non-nuclear application areas?
- What are the criteria for identifying successful performance adaptations?
- What specifications should a method designed to support learning from successful performance adaptations fulfil, i.e., how should the current operating experience practices, e.g. event investigations, be developed in order to facilitate learning from successful actions and decisions?
- How should successful performance be communicated to promote the transfer of lessons learned between plants/units?

2.1.3 Specific research results of WP1

- Better understanding of the role of successful events as a mean for promoting nuclear safety
- Method for analysing successful aspects of human and organisational performance
- Guidance on how to share the lessons learned from successful performance adaptations within the NPP and among other NPPs
- Better understanding of the needs for development in the operating experience feedback system.

Partners and person months allocated to WP1 to be given in the table.

<table>
<thead>
<tr>
<th>Partner in WP1</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>3.0</td>
</tr>
</tbody>
</table>

2.1.4 Task 1 (T1.1) Methodology development

2.1.4.1 Progress in 2015-16

In 2015, we carried out a literature review that included more than 50 scientific articles to identify theories that define success. The focus was on leadership and project management literature, and literature from safety
science was also reviewed where possible. Based on the literature review, success was defined as matching, exceeding or returning to an expected performance level. In order to capture successes, a range of metrics was also identified. These were associated with processes, project management (i.e., cost, budget, and scope), product acceptance, organization (e.g., creating financial, technological, social or structural benefit for the organization), preparation for future scenarios, socio-affective impact, as well as lagging and leading safety metrics.

Some of the main findings in our literature review was that success, especially as it is related to learning, is more than merely achieving a pre-specified task outcome. Success also contains such elements as social construction (i.e., success is a result of processes like interpretation, sense-making, and subjective evaluation), criteria dynamics (the criteria against which success is assessed against change over time), outcome dynamics (task outcome itself changes over time – what is initially thought of as success later becomes failure or vice versa), hierarchy and stakeholder multiplicity (different stakeholders at different levels of the organisation see success from different perspectives allowing them to assess task outcome against different types of criteria), and process boundaries (the existence of boundaries regarding how task outcome is achieved) – this is especially relevant in safety-critical organisations where just performing a task regardless of the means used to achieve it might not be acceptable.

The literature study further revealed that learning from successes differs from learning from failures: successes often attract less attention than failures and are less likely to motivate extensive organization-wide changes; lessons learned from successes appear to rely more on less formal knowledge storage, while failures are more often stored in stable organizational structures; and successes tend to reinforce existing knowledge while failures challenge it.

The findings from the literature review were used to design the empirical studies and served as a starting point for developing the principles for identifying successful actions and decisions in NPP context.

In addition to reviewing the literature, a preliminary framework for learning from successes was developed for the project intermediate report (Viljanen et al., 2016a). This work was further elaborated during 2016 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.

2.1.4.2 Year 2017

**Goal.** The development of pilottable tools to enable learning from success

**Content.** We will choose a selection of tools that are either already in use in the nuclear industry or that are easily pilottable in practice, and integrate a success perspective into them. This work will be guided by the principles and the process described in the guideline developed earlier in this project. The tools to be developed relate to three types of success: recovery successes (e.g. augmenting event investigation methods with a success analysis), normal successes (e.g. development of observation techniques to identify successes in daily work) and extraordinary successes (e.g. improving the sharing of lessons learned from successes). The tools to be developed will be selected jointly with case study organizations.

**Methods.** Researcher workshops, literature review into the selected tools

**Deliverables.** A set of practicable tools to be piloted in T1.3

2.1.5 Task 2 (T1.2) Empirical studies on learning from successes

2.1.5.1 Progress in 2015-16

In 2015 we carried out case studies in one Swedish (Ringhals) and one Finnish (Fortum Loviisa) NPP. The case study in Fortum Loviisa consisted of two approaches to data collection: one that was based on document analysis of incident reports and another based on field observations.

In document analysis of the incident reports we attempted to identify how successful decisions and actions can be identified from data that is essentially structured around a failure. Based on our understanding of the potential principles for capturing successful events, we chose a selection of reports which we discussed with relevant NPP personnel to gain more detailed insight regarding the course of events and possible success contributors and lessons learned.

The field observations done at Fortum Loviisa were carried out during the 2015 outage. We were especially focused on heavy lifting tasks, but smaller tasks and general activities at the plant were also observed. During almost all of the task observations, two researchers were present to ensure sufficient diversity and reliability. Observations were done in both reactor and turbine halls. Our approach to the observations was generally exploratory.
– we collected and noted any activity that took place during the observations, with a special focus on various types of adaptations. Brief interviews with field-workers were also conducted when possible. After the outage we carried out more extensive interviews with key personnel regarding some of the major tasks we observed.

The preliminary findings (described in detail in Viitanen et al., 2016a; Viitanen et al., 2016b) helped us to identify the various types of successes that occur in maintenance activities and the factors that contributed to these successes. We found that in addition to failures, successful experiences also appear to trigger learning, albeit usually more local and informal – the lessons learned seemed to remain within work teams or individuals. Often, however, successes were found mundane and largely ignored; respectively, in incident situations, failures took focus even if successful actions or decisions took place. Learning from successes appeared to be mostly unsystematic, although there were some attempts to formalize it in case organizations. We also found that there is a relative lack of tools for learning from successes and that there are other methodological issues (e.g. socio-cultural acceptability) that need to be considered when promoting learning from success.

The work will possibly continue in 2018.

2.1.6 Task 3 (T1.3) Initial feedback, methodology validation and dissemination of results

2.1.6.1 Progress in 2016

In 2016, we aimed to provide the representatives of the case study NPPs an opportunity to reflect our findings from T1.1. and T1.2. in a workshop. The main topic was to discuss the practical potential of the principles for capturing successful events in operating experience work. In addition, the workshop 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. In 2016, the piloting and validation activities were halted due to the cancellation of NKS funding.

2.1.6.2 Year 2017

**Goal.** Pilotling the tools for enabling learning from successes and disseminating the overall findings of the project to nuclear industry practitioners and the scientific community.

**Content.** First, the tools that have been developed during the project will be piloted in collaboration with the case NPP organisations in Finland (Fortum Loviisa) and Sweden (Ringhals). Moreover, discussions with Fennovoima are ongoing regarding their participation as an additional case organization for the piloting. In addition, a document-based piloting will be carried out in which the guideline will be applied to create lessons learned from the successes achieved in selected aspects of the Fukushima 2011 accident. After the piloting, modifications may be made to the tools and the guideline. Secondly, co-operation with CORE WP6 will be arranged (in March - April 2017) to gain new insight in developing the guideline and the practical tools. Thirdly, a scientific publication (conference paper) will be written to present the project findings to the scientific community. Finally, project findings will be disseminated and discussed in an international workshop with nuclear industry practitioners (at HUSC seminar to be held in 7.-8.11.2017).

**Methods.** At least two workshops with operating experience personnel at case study organizations; feedback analysis from workshops; document analysis; researcher workshop with WP6; international nuclear industry practitioner workshop.

**Deliverables.** 1) scientific publication (conference paper) jointly with T1.1; 2) Presentation and workshop to be held at HUSC seminar in 7.-8.11.2017

2.1.6.3 Year 2018

In 2018 the plan is to continue piloting the developed tools and write a report describing the practical tools and the results from piloting. Furthermore, we aim to disseminate the results to the scientific community, for example, by revising the conference papers and other publications written during the earlier phases of the project into peer-reviewed journal articles.
2.2 Work package 2 (WP2) Developing work-based learning in the NPP domain (WOBLE-NPP)

The successfulness of NPP operator crews in demanding situations can be promoted by supporting certain kinds of work practices. Upholding accurate situation awareness by reciprocal communication, careful consideration of different information sources, and anticipation of events exemplify positive features in work practices. These types of work practices would go hand-in-hand with a work orientation emphasizing interpretativeness instead of mere ‘mechanical’ actualizing of pre-determined procedures. Assumedly, work practices comprising these features promote resilience as well as continuous learning – the latter benefit is understandable, since as discussion and interpretation takes place, more profound understanding of the system and positive modifications to the existing work practices emerge.

A three tier categorization is explicative of what is meant with ‘interpretative’ here. The categories, which were originally generated by Norros (2004), have been labelled as 1) interpretative, 2) confirmative, and 3) reactive practice (see Table 5 below). Behaviours belonging to the interpretative class can be exemplified by crews gathering redundant and diverse information before conducting process interventions: they consider different types of information, such as alarms, display support systems, parameter values, trends, and automation information thus portraying a strive for profound understanding of the actual process situation. Confirmative practice, in turn, can be exemplified by the behaviour of double-checking information in a rule-following manner. Finally, the reactive practice is characterized by utilization of alarm information as the only basis for triggering behaviour.

Table 5. Features and findings on NPP operator work in three categories of work orientation.

<table>
<thead>
<tr>
<th>Category</th>
<th>Reactive</th>
<th>Confirmative</th>
<th>Interpretative</th>
</tr>
</thead>
<tbody>
<tr>
<td>Descriptive feature of work</td>
<td>Operator work perceived as ‘mechanistic’</td>
<td>Rule-following is emphasized</td>
<td>Comprehension and intelligent use of procedures is emphasized</td>
</tr>
<tr>
<td>orientation</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Features of work practices</td>
<td>Activity based on the immediate features of situation (e.g. alarms) and no active anticipation.</td>
<td>Selecting and considering pre-determined courses of action; double-checking of information; trend information guides activity and is considered in addition to procedures and alarms</td>
<td>Anticipation of events; dialogue within the team; active ‘sense-making’ (e.g. redundant information gathering); striving for profound understanding of the system</td>
</tr>
<tr>
<td>Findings</td>
<td>27% of operator activity can be considered as reactive¹</td>
<td>41% of operator activity¹; the most common orientation to work in operator interviews²</td>
<td>32% of operator activity¹; interpretativeness appreciated by NPP procedure designers³</td>
</tr>
</tbody>
</table>

¹ Savioja et al. 2014
² Norros et al. 2014
³ Wahlström et al. 2014

2.2.1 Specific aim of WP2

The aim is to create a development method for helping NPP operators to be more interpretative in work. Since it is unclear how to promote interpretative work practices and the interpretative orientation to work, the process of learning interpretative way of working needs to be investigated. Collaboration between FIOH and VTT is justified, since both VTT and FIOH have a long history in workplace developmental interventions (Laarni et al., 2011; Leppänen, 2001; Leppänen et al. 2008; Norros & Klemola, 1999; Schaupp 2011; Seppänen et al. 2009).

2.2.2 Specific research questions of WP2

The main research question is how to promote interpretative practice among NPP operating crews. This question implies a creation of a development method that promotes the interpretative work orientation. To address this question, the following auxiliary research questions are needed:

1. What kind of developmental method could be created based on the idea of interpretative work?
Since the aim is to provide a permanent change for operator work practices, also the following research question is addressed:

2. How to integrate the development method to existing operator training and NPP training developer work?

The basic hypothesis of WP2 is that these goals can only be achieved in close collaboration with the operating organizations.

2.2.3 Specific research results of WP2

A new development and training method will be developed, which is of practical value for power companies in Finland and also internationally. The new development method will be based on existing approaches for developing work, such as Activity Clinic (Clot, 2011; Sannino, 2011) and Change Laboratory (Virkkunen and Newnham, 2013). Drawing from these approaches, the method will include dialogical consideration of existing work practices; these discussions, including operators, trainers and researches in varying compositions, will be stimulated with video data on problematic and challenging work situations (in a control room simulator) and researchers’ findings on challenges and built-in contradictions in the work. In other words, the operators will be able to consider their work practices individually and in groups. However, in contrast to Activity Clinic and Change Laboratory methods, the specific feature of the WOBLE-NPP development method will be, first, viewing work practices through the concept of interpretative practice and the theoretical three-tier categorization of practices (Table 5). This, in turn, implies the promotion of resilience as well as sustainable change within the organization as, assumedly, the interpretative orientation to work promotes learning. In other words, the aim of the development method is to foster learners’ ability to learn. To further promote a sustainable change, the WOBLE-NPP method involves a close collaboration with the trainers and training developers working for the studied organization. The intention is that the development method will become part of the applied training repertoire in one form or another. For example, with the new method, the trainers of NPP operators will be more equipped to evaluate performance in simulator situations; by considering the interpretativeness of work practices, they will have more varied tools for considering the quality of simulator activity.

A positive stance on developing a new method for training has been indicated by power companies; our study will mainly be conducted at Fortum, but the results will also be presented at TVO. FIOH and VTT already collaborate in an existing Academy of Finland study called “Interpretative work: Developing new forms of work-based learning for the age of digitalisation” (WOBLE) in which interpretativeness is studied in robotic surgery. WOBLE-NPP is a direct continuation to WOBLE. In other words, the basic findings on the development method created in WOBLE will be applied in WOBLE-NPP. Both WOBLE-NPP and WOBLE are provided with intellectual support and guidance by Pascal Béguin (Professor of Ergonomics at the Université Lumière, Lyon 2), Laure Kloetzer (Assistant Professor in Psychology at the University of Neuchâtel) and Harry Daniels (Professor of Education at the University of Oxford).

The research results have general relevance in view of the updated SAFIR 2018 call (Update to the Framework plan for 2017 call): increased capability to interpret situations from various angles and manners promotes resolution of new presently unknown threats and operation of the plant in the load following mode.

Partners and person months allocated to WP2 to be given in the table.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>2.0</td>
</tr>
<tr>
<td>FIOH</td>
<td>0.7</td>
</tr>
</tbody>
</table>

Since the titles and contents of the tasks have slightly changed, the progress of all tasks in 2016 is presented first, and after that the future work in each task is described.

2.2.4 Progress in 2015-16

2.2.4.1 Task 1 (T2.1) Creation of development options

The goal of the first year (2015) was to provide the basis for the accomplishment of a development method with the NPP operators. The initial idea was that this would involve the collection of comparison data for assessing the impact of the development method. However, upon initial interviews it came out that the operating organization possesses several practical and administrative restrictions for taking new training methods into use and, to some
degree, for data collection – i.e., it was found that it would be very important that the new method is in par with the needs and requirements of the studied organization. In practice, as there is limited space in the existing training curriculum for new kind of training, the new training method should enhance the existing training by introducing new features to it. Therefore, and in view that simulator session content is largely determined by the aims of the operating organisation, instead of trying to achieve objective comparison data on operator behaviour, the focus was directed to identify training needs as well as challenges and shortcomings in the current training practices.

In 2016, a development method was created. 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.

A conference article has been completed jointly with T2.2 and T2.3.

2.2.4.2 Task 2 (T2.2) Training method options evaluation

The aim was to select and test the best training method options. In Task 2, the ideas generated in Task 1 were discussed and tested. 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). As of writing, 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.

The method prompted the operators to consider their own simulator performance in view of criteria of good work as established by the operating organization (these are very much in line with the criteria of interpretive practice, as presented on Table 4). Additionally, the operators were prompted to discuss 1) psychological stress (because stress is one of the themes of CORE), 2) the aims of the simulator training session, 3) challenges within the session and 4) new learning. The discussions that emerged after simulator sessions were rich in content in considering 1) work practices, 2) procedure format and social interaction, 3) plant dynamics, and 4) psychological stress. The group reflection involved in our method could also be seen as an opportunity to defuse, that is, to attain a feeling of normality after a stressful event by discussion.

A conference article has been completed jointly with T2.1 and T2.3.

2.2.4.3 Task 3 (T2.3) Creating and studying intra-organisational understanding on operators’ training needs

The aim was to increase and study understanding on operators' training needs within the operating organization, in view of findings made in 2015. In view of the operating organization under study, as of writing, this work is still in progress. However, related to the task the following actions were conducted:

- Training needs were discussed with the operating organisation for developing a training method.
- Training needs were discussed with another operating organisation for developing training.
- Training needs were discussed with the regulator.
- The findings on operators' training needs were used in finalizing a scientific article on NPP HR developers work (initiated in SAFIR 2014 by participating researchers) (in progress).

A conference article has been completed jointly with T2.1 and T2.2.

2.2.5 Year 2017

2.2.5.1 Task 1 (T2.1) Evaluating and developing the self-evaluation method

In 2017, the development method will be further evaluated, refined and established. Based on the initial tryouts in 2016, the evaluation method seems to be beneficial. However, some open questions remain: 1) how can the self-evaluation method promote exchange of good practices between the operator shifts?; 2) are there elements in the method that are unnecessary or are new elements needed (from the point of view of resilience, in particular)?; 3) how often and after what kind of simulator training should the self-evaluation method be applied?

Goal. To evaluate the created self-evaluation method (for evaluation within shifts and individually).
Content. Observation and analysis of the use of the method; discussions with operators, trainers and training developers.

Method. Analysis of the method use; collecting data on operators’ perspectives on the method; workshop sessions (integrated to operator training).

Deliverable. A conference article on developing training in NPP (together with T2.2; to be submitted by the end of September of 2017).

2.2.5.2 Task 2 (T2.2) Creating and studying intra- and inter-organisational understanding on operators’ training needs and contrasting these needs to the existing NPP training instructions

Overall, our study has generated understanding of the training practices, training needs and characteristics of good training. Then, for making a positive real-life effect on the operating organizations, it has to be studied what the organisational conditions and possibilities for developing NPP training instructions are in practice and discuss these ideas within different operating organisations and at different levels of organizational hierarchy. Furthermore, given the existing studies of WP2, we are adept to consider the regulations on training, that is, whether they actually support formative development or whether there is a need for a change in regulations as well – these considerations will be discussed with the regulator.

Goal. The aim is to exchange the views on training needs and good training with the operating organisations and the regulator, in view of findings made in 2015 and 2016 (here also considering the existing regulations).

Content. Interviews and workshops with relevant organizational units at the operating organizations to discuss training needs.

Method. Interviews.

Deliverable. A conference article on developing training in NPP (together with T2.1; to be submitted by the end of September of 2017).

2.2.6 Year 2018

In 2018, the aim is that the new development method or training modification would be a part of the competence development practices at the Finnish NPPs. Thus, the learning sessions would be conducted by in-house developers and NPP operator trainers, but they would also be provided with assistance. If needed, the method would be further adjusted, thus further enhancing the method and ensuring its practical usefulness. Furthermore, we have stimulated discussions on the existing NPP operating training guidelines. An additional task would be to disseminate the overall lessons-learnt to potential international partners.

2.3 Work package 3 (WP3) Supporting operational resilience in complex and dynamic environments

2.3.1 Topic 1: Multitasking and management of goal conflicts

The rate of interruptions has drastically increased during the last decades in many work domains. A survey conducted in Germany showed that the rate of interruptions has doubled in the past twenty years and has become of one of the factors that cause a lot of stress (BAuA, 2013). At the same time, multitasking is growing steadily, and multitasking has become a normal condition in many work domains so that it is even more difficult for workers to recognize that they are multitasking. Multitasking, interruptions and distractions are also a natural part of operator and maintenance personnel work in nuclear power plants, and operators and technicians are experts at managing task switching and interruptions/distractions. Interruptions are advantageous, if they provide critical information to the operators and technicians. Sometimes interruptions, however, cause problems in their work. Efficient management of interruptions is required in many tasks and domains (for a review, see Salvucci & Taatgen 2010), and task switching has been extensively studied in such domains as health care (for a review, see Hopkinson et al., 2013) and piloting an aircraft (Parasuraman et al., 2000; Wickens, 2002). However, multitasking has not been studied in
process control work. Multitasking is an important issue to study, since control-room operators are frequently encountered by multiple task activities that they have to perform simultaneously or in rapid succession. Knowledge is needed on one hand of causes of interruptions and of their impact on cognitive load, and on the other hand of how operators manage task switching and parallel work at full power operation, in load following mode and in incident and accident conditions.

Management of multiple task threads simultaneously typically requires distributed cognition which refers to the notion that knowledge lies not only in human mind, but it is distributed over his/her social and physical environment (e.g., Hutchins, 1995). Because of the complexity of situations and of the richness of the information, modelling of distributed cognition is challenging. Typically, the collected ethnographic data has been analysed by describing qualitatively, e.g., communication patterns and decision making processes in complex work tasks (e.g., Hazlehurst et al., 2007). Recently, it has been suggested that network models such as the Event Analysis of Systemic Teamwork (EAST) method could be used in the analysis and modelling of distributed cognition in complex systems (Stanton, 2013; Stanton et al., 2008). The EAST method has been developed to investigate the work of distributed teams in complex socio-technical systems (Stanton et al., 2005), and it is based on a network of networks approach, according to which task, social and information networks have to be brought together into the common analysis framework.

Multitasking in operator and maintenance personnel work is a direct consequence of goal conflicts that are characteristic to the work in industrial organizations (Dekker, 2011; Woods, 2006). They have to cope with multiple and often interacting and conflicting goals in their work. Even though incompatible goals typically come about at the organizational level, they have to be mastered at the operating personnel level. They are derived, for example, from management policies, regulatory guidelines and economical pressures. For example, if the power plant will be started to be operated in the load following mode, additional goal conflicts may emerge, and extra cognitive load may be put on operative personnel. An important research question is what types of goal conflicts operator and maintenance personnel meets in their work and how goal conflicts are manifested and managed at the operational level. The first research aim is to identify the conflicting goals, trade-offs and double-bind situations that they have to master in their daily work and try to trace back their underlying causes. Also, possible challenges and conflicts caused by the new ways to maintain power balance will be addressed. The second aim is to identify the multiple strategies people use in trying to cope with multiple goals and trade-offs.

Since we do not have much knowledge about operative personnel’s work conditions and their work practices in different plant conditions (e.g., at normal power operation, operation in the load following mode and during outages), this kind of information has to be collected by various methods. Ethnographic observations would provide the most direct evidence of work conditions and practices, but unfortunately outside observers are not allowed to enter power plants at normal power operation. Therefore, data must be collected by more indirect methods such interviews and surveys. Our first aim is to conduct some interviews to get an overview of the role of multitasking, interruption management and mastering conflicts in work of operative personnel. Based on these interviews, an online survey will be developed and deployed to the operative personnel of the Finnish NPPs.

2.3.2 Topic 2. Diagnostic reasoning and problem solving in operator work

There is evidence that control room operators have problems in diagnosing complicated events and multiple simultaneous events in process industry (e.g., Kim et al., 1999; Rouse, 1978; Toms & Patrick, 1987; Wohl, 1982). Even though there is a lot of research on finding faults in complex incident situations, there is quite little knowledge of the problem solver’s cognitive strategies, states and activities in troubleshooting situations (see, however, Bereiter & Miller, 1989; Hoc & Carlier, 2000; Patrick, 1999). Also, quite little is known about the effect of stress on troubleshooting performance in industrial domains, even though it is well known that high stress has a detrimental effect of cognitive performance such as reasoning, spatial cognition and memory (e.g., Harris et al., 2008). This knowledge is, however, important in modelling diagnostic reasoning and in developing training interventions and decision support systems to support reasoning and problem solving in industrial settings. One possible approach to illustrate the problem solver’s mental states and activities is based on the problem behaviour graph described by Newell and Simon (1972) which represents a person’s successive knowledge states transformed by information processing activities. Another similar notation system is MAPS (Mental State and Activities in the Problem Space) that has been developed by Patrick (1999). These notation systems are, however, based on a quite simplified information processing view of human cognition. Therefore, these approaches have to be enriched to better take into account the complex dynamic context in which cognitive processes take place (Norros, 2014). In addition, these notation systems entirely focus on problem solving at the individual level, even though in a NPP control room problems are typically solved at the team level together with other operators. The above mentioned EAST method provides a practical tool for the analysis of troubleshooting and complex decision making at the operator crew level.
Solving complex diagnostic problems is an example on knowledge-based information processing in which the problem solving is carried out in a conscious and rational way (e.g., Rasmussen, 1983). Another mode of task execution is skill-based processing which is characterized as intuitive, automatic, fast, effortless and unconscious. Skill-based processing is to a large extent based on mental maps, rules of thumbs and tacit knowledge (Croskerry, 2009). According to the claim of bounded rationality (Simon, 1978), since the amount of relevant information is in many everyday situations enormous, the decision maker is not able to process it all, and he/she is prone to drive in skill-based information processing even in situations in which a more thoughtful processing would be preferable. Therefore, people often fail to process information that is easily available to them, and they may be prone to many types of cognitive biases and errors. However, from the resilience point of view, people’s actions are rationale taken into account their goals, knowledge and focus of attention at the time (Hollnagel, 2006; 2011a). In order to understand people’s behaviour, we therefore have to study how limited knowledge and mindset and how multiple interacting goals shape the behaviour of people in dynamic situations. Heuristics and rules of thumb control room operators use in their daily work can be considered as context-specific resilience skills, strategies and competencies (e.g., Furniss et al., 2011). However, even though it would be important to explore which kind of heuristics and other kinds of adaptive strategies control room operators bring to bear in dynamic problem solving and decision making situations, there is very little research on cognitive heuristics that operators use and biases that they commit in control room settings.

It is difficult to study non-analytical, implicit reasoning which is based on rapid recruitment of cognitive heuristics and rules of thumb. If ethnographic methods are applied, we have to rely on individuals’ observable behaviour, on the basis of which it is difficult to make any inferences about their thought processes. On the other hand, indirect methods such as interviews and questionnaires are also problematic, because individuals have not necessarily conscious access to implicit reasoning processes, and they are not able to voluntarily control them. One solution is to obtain data of cognitive heuristics both through observational and indirect methods. In any case we have to carefully specify which kinds of behaviours we are looking for, and by which way cognitive heuristics are manifested in behaviour.

2.3.3 Specific aims of WP3 in 2017-18

In 2015, we first studied the impact of multitasking and interruptions/distractions on work performance in safety-critical domains by preparing a literature review. Second, a method was developed for the identification and classification of distractions in operator work, and it was tested in the analysis of simulated accidents. Third, functional situational models of the TVO Olkiluoto simulation runs were prepared, and they were used in the analysis of simulation test data. Fourth, a literature review on troubleshooting in operative work was prepared, including a description of the development of MAPS-based notation system for troubleshooting modelling.

In 2016, 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. In addition, the Functional Situation Model (FSM) method (e.g., Savioja et al., 2012) has been further developed.

In 2017-18, the aim is to investigate multitasking and interruption handling among CR operators, field operators and maintenance personnel in normal plant states and during outages. In order to obtain a general overview of multitasking in operative work, we will conduct some interviews with operators, trainers and maintenance personnel. Based on these interviews, the draft version of the survey will be developed, and it will be discussed with contact persons in the NPPs. The survey will address the following issues: 1) what kind of implicit communication etiquette and rules can be identified that guide interruption and distraction management at CR level, 2) how operators manage parallel tasks, task switching and interruptions in different plant states (i.e., normal operation, load following mode, incident and accident conditions, and outage), 3) what is the impact of interruptions and distractions on operators’ cognitive load and stress, 4) what are the critical goal conflicts, trade-offs and double-binds that operator and maintenance personnel have to meet in their daily work, and 5) what are the possible challenges and conflicts related to the operation of an NPP in the load following mode.

Results of the analysis and modelling of multitasking and goal conflict management are used in the development of guidance for better management of task switching, interruptions and goal trade-offs in process control work in various plant states (i.e., normal operation, load following mode, incident and accident conditions and outages).

In the second task the aim is first to further develop the modelling tool developed for the analysis of collaborative troubleshooting and apply it to the analysis of video and process tracing data from fault-finding scenarios. Second, based on the analysis of video recordings of simulated incident and accident scenario runs and process tracing interviews, the aim is identify and characterize typical cognitive heuristics, biases and rules of thumb operators bring to bear in different operating conditions. The outputs of the above-mentioned research work are used in developing a training material/programme on advancement of diagnostic reasoning skills in control room settings in 2018. In
addition, a concept of a computer-based learning environment for training troubleshooting will be outlined in 2018 (see, e.g., Jonassen, 2011).

2.3.4 Specific research questions of WP3 in 2017-18

- What strategies operators use in managing interruptions/distractions and task switches in various plant states (e.g., in normal power operation, in the load following mode and during outages)?
- What impact interruptions and distractions have on operators’ cognitive load?
- What goal conflicts operators meet in their daily work and how they cope with them?
- What kind of modelling techniques are needed to characterize distributed cognition in control room settings?
- How to further develop the Functional Situation Model method so as to permit better analysis of operator behaviour in incident and accident situations?
- What diagnostic difficulties operators meet in trying to solve complex faults and how they try to tackle these difficulties both at the individual and team level?
- How various types of cognitive heuristics, biases and rules of thumb manifest themselves in operator work and what effects their usage have?

2.3.5 Specific research results of WP3 in 2017-18

- New knowledge about multitasking challenges in operator work
- Analyses of distributed cognition in control room environments
- Revised version of the Functional Situation Modelling technique
- Better understanding of how goal conflicts and trade-offs are managed at the operational level in various plant states
- New knowledge of operators’ cognitive states and activities in complex troubleshooting situations
- Training material and/or programme on advancement of diagnostic reasoning skills
- Concept of a computer-based learning environment for troubleshooting training
- Better understanding of how cognitive heuristics and rules of thumbs are manifested in control room work

Partners and person months allocated to WP3 to be given in the table.

<table>
<thead>
<tr>
<th>Partner in WP3</th>
<th>Person months in 2016</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>2.4</td>
</tr>
</tbody>
</table>

2.3.6 Task 1 (T3.1) Multitasking and management of goal conflicts

2.3.6.1 Progress in 2015-16

A literature review on multitasking and the impact of interruptions/distractions on work performance in safety-critical domains titled “I am too busy to think – Multitasking and interruptions in safety-critical domains. Literature review” was completed. According to the review, interruptions and distractions that divert attention away from the task at hand to another task have typically a detrimental effect on performance. However, if the distraction is related to the primary task and add new information that is helpful in task execution, its effect can be positive. Even in the latter case the effect may be negative, if the information is not directly associated with the task the worker is just performing. According to the review, there is a lot of research on how to reduce the negative effects of interruptions and multitasking, but it is important that the remediations do not increase cognitive burden and lead to additional drop in performance. A conference paper titled “Multitasking and interruption management in control room operator work during simulated accidents” was published in 2016.

A review on work modelling tools (e.g., social network analysis, information/knowledge/propositional network analysis and time-window sequential analysis) has been prepared in which the modelling tools have been analysed by several criteria (e.g., soundness of theoretical basis, applicability to the analysis of troubleshooting in nuclear
domain, availability of suitable software, labour and resource intensity, and ability to differentiate good and bad performance). The results suggest that some modelling tools, such as social network analysis and information network analysis, are valuable in analysing information sharing and team situation awareness in collaborative troubleshooting, but they do not provide detailed information about the sequential evolution of a team’s knowledge state throughout the diagnostic process. On the other hand, the tools that represent an individual troubleshooter’s successive knowledge states which are transformed by information processing activities are totally ignorant of the distributed nature of complex troubleshooting. Therefore, these approaches have to be tailored to better suit the analysis of team collaboration and co-operation in incident and accident situations.

We have analysed operator practices in simulator conditions with the FSM method (e.g., Savioja et al., 2012), 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 (see WP4: Supporting operator performance in extreme stress) from operator activity by synching annotations to HRV data on a timeline.

2.3.6.2 Year 2017

**Goal.** The aim will be to
- interview instructors, operators and/or maintenance personnel on multitasking, interruption management and manifestation of goal conflicts in various plant states
- outline a theoretical framework on multitasking and interruption management in operative work
- develop and pretest a questionnaire on multitasking, and interruption and goal-conflict management

**Content.** We will analyse in a more detailed fashion which cognitive strategies operators use in the management of multiple parallel task threads in normal process control or, e.g., in load following mode or during an outage. The aim is also to investigate by which way goal conflicts and trade-offs are manifested and managed at the level of operating personnel in NPPs by interviewing operators, field workers and maintenance personnel. We will outline a theoretical framework on multitasking and interruption management in operative work in NPPs, and develop an online questionnaire on the topic. The questionnaire will be pretested by a small group of operative personnel in 2017. The survey itself will be launched in the beginning of 2018.

In addition, a more detailed performance modelling of the simulator data, collected jointly with WP4, will be conducted to support the analysis work in T4.1.

**Method.** Interviews, modelling and survey design.

**Deliverable.** A slide set on multitasking modelling and a preliminary version of the multitasking survey (by the end of 2017).

2.3.6.3 Year 2018

In 2018, the online survey will be launched, and its results will be analysed and published. Based on the survey results, guidance is developed for better management of task switching, interruptions and goal trade-offs in process control and maintenance work.

2.3.7 Task 2 (T3.2) Diagnostic reasoning and problem solving in operator work

2.3.7.1 Progress in 2015-16

Functional situational models of the TVO Olkiluoto simulation runs were prepared, and they will be used in the analysis of simulation data (for details, see 2.4.7.1). A literature review on troubleshooting in operator/maintenance personnel work was prepared, including a description of the development of MAPS-based notation system for troubleshooting modelling.

We have outlined the first version of a modelling approach suitable for analysing collaborative diagnostic reasoning and troubleshooting of a NPP control room crew which is based on existing methods and tools. The approach describes the progress and evolution of a CR operator crew’s knowledge states throughout the critical sections of
a simulator run. We present 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.

A conference paper titled “Modelling collaborative troubleshooting in nuclear domain” has been prepared and will be submitted to ANS NPIC&HMIT 2017 conference to be held in San Francisco in June 2017.

2.3.7.2 Year 2017

**Goals.** Our aim is to
- further develop the modelling tool and apply it to the analysis of complex troubleshooting scenario run.
- based on video recordings of simulated incident and accident scenarios and their debriefing sessions, the aim is identify and characterize typical cognitive heuristics, biases and rules of thumb operators bring to bear in these conditions.

**Content.** The aim is to acquire a better understanding of the mental processes and team-level collaboration in fault finding and in diagnosing faults. We will analyse simulator data on troubleshooting in complex fault states by using the notation system developed in 2016. In addition, cognitive heuristics, biases and rules of thumb are collected by analysing video material of simulator runs and debriefing interview sessions that has been collected in simulator settings both at the Fortum Loviisa NPP and TVO Olkiluoto NPP.

**Methods.** Video analysis; modelling.

**Deliverables.** Two conference articles, one on modelling of troubleshooting activities and the other one on cognitive heuristics, biases and rules of thumb in operator work (both by the end of 2017).

2.3.7.3 Year 2018

In 2018 we will interview simulator trainers and operators in order to get a better understanding of cognitive heuristics and rules of thumbs that operators use in their work and to get a clearer perception of which way these heuristics affect their work performance in various plant conditions. Training material and/or a training programme on advancement of diagnostic reasoning skills in operative work is prepared and tested as a part of the annual refresh training. Additionally, a concept of a computerized learning environment for troubleshooting training will be outlined.

2.4 **Work package 4 (WP4) Supporting operator performance in extreme stress**

2.4.1 **Topic 1. Objective measurement of acute stress response in simulated risk situations**

In order to efficiently improve power plant workers problem solving and performance especially under high pressure, more accurate information on the stress levels and critical stress-inducing factors in (simulated) risk situations is needed. Typically, the evaluation of workload and stress is conducted by collecting self-reports of experienced stress. Sometimes, this provides an accurate enough representation, but often there is a substantial risk that the subjective perception is biased (Sallinen et al., 2004; 2008; Haavisto et al., 2010). Another major limitation of these subjective measures is that the operator can often only assess the overall experience of workload of activities but cannot reflect changes in workload during the execution of activities. Moreover, collecting self-reports during a task has a disrupting effect on task performance. Thus, objective, non-invasive and unobtrusive measurements of workload and stress levels are needed to provide more accurate information on the state of the individual at desired time, without disturbing the primary task.

Acute stress affects performance by modulating human information processing roughly according to an inverted or quadratic U-shaped function (for a review, see Lupien et al., 2007). This means that low-to-moderate amounts of the stress-released hormones glucocorticoids, catecholamines and noradrenaline can improve cognition and performance, while the higher doses tend to impair them. With increased levels of stress, sensory processing *per se* may become improved (de Kloet et al., 2005; van Marle et al., 2009), but the focus of attention narrows (tunnel vision; Christianson, 1992), person experiences difficulties in directing attention to relevant information (Tanji & Hoshi, 2008; Henderson et al., 2012) and becomes increasingly distractible by irrelevant information (Skosnik et al., 2000; Braunstein-Bercovitz et al., 2001; Aston-Jones & Cohen, 2005). Also memory performance, especially
memory retrieval, declines (e.g., Dominique et al., 2000; Kuhlmann et al., 2005a; 2005b; Smeets et al., 2006; 2008). In practice, this means that the problem solving, learning, and executive functions, such as finding the correct emergency operating procedure (EOP) in minimal time, and acting according to the EOP can become compromised, especially in the most stressful, i.e. the most critical, highest-risk accident and incident situations.

This study uses a multi-method approach that combines the quantitative continuous physiological measurements [electrocardiography (ECG) and heart rate variability (HRV), electrodermal activity (EDA) and accelometry, qualitative field study methods (observations, modelling of the simulations), interviews and survey methods (e.g., NASA-TLX and KSS). The physiological data is compared with the self-reports, simulation trainer evaluation, and performance scores in order to obtain a coherent view on the effects of workload and stress on performance. The existing literature indicates that physiological recordings serve as good predictors of workload and performance also in nuclear power plant environments (Hwang et al., 2008; 2009; Gao, 2013), but the recordings need to be accommodated and tailored to meet local practices and more advanced technologies. Novel gaze-trackers provide reliable measures of gaze direction and gaze duration, and can be used to model information seeking and processing, providing information on the operators’ locus of attention and processing strategies. As the technologies advance with high speed, they need to be tested for appropriateness, validity, reliability and intrusiveness. Moreover, the functional significance of the stress response in the local work context needs to be evaluated, i.e., the effect of the stress response on the actual performance of the power plant operator in the simulated incident and accident situations will be quantified.

2.4.2 Topic 2. Stress management within the context of nuclear power plant operator work by developing operator and instructor training

While slight stress improves and excessive stress impairs performance, stress management can be used to enhance performance in demanding situations, such as nuclear plant incidents and accidents as well as in training. According to recent research, understanding, i.e., “cognitive appraisal” of the acute stress response, can have a critical effect on both the nature and adaptivity of the physiological response and performance level of the individual under stress (Jamieson et al., 2010; 2012). In these studies, participants who were given information about role of the acute stress reaction as an adaptation to meet the demands of a potentially threatening situation had a more desirable and healthier, approach-type pattern of physiological stress (stronger heart output, but lower vasoconstriction) than the controls, or those who were instructed to ignore the stress reaction (a typical, often intuitive reaction to stress; Jamieson, 2012). Importantly, those participants interpreting the stress as supporting factor also performed better in test situations (Jamieson et al., 2010). Understanding of the stress physiology, and conveying this information to the NPP crew can thus improve the performance level and reduce the undesirable aspects of stress at work.

Simulator trainers need theoretical knowledge and practical skills on a diverse set of topics and reflective skills to monitor and assess their own work (Leinonen, 2009). Six key competence areas were identified in Laamm et al.’s (2011) study: 1) Profound familiarity and understanding of the nuclear power process and plant systems; 2) profound knowledge and control of the simulator computer; 3) basic knowledge of Human Factors and human performance; 4) skills to develop and design training programs based on the Systematic Approach to Training; 5) pedagogical abilities; and 6) commitment to the continuous maintenance and growth of professional ability. Specifically, our own teaching experiences suggest that simulator trainers consider it important to learn more about the effects on stress on operator performance, and stress management, but there is little knowledge about what kind of training would be most important and most helpful to them.

2.4.3 Specific aims of WP4

Objective physiological recordings of stress markers (heart rate variability, HRV; electrodermal activity, EDA) are applied to quantify the stress and workload of power plant operators in (simulated) risk situations. The effects of workload and stress on operator performance in these situations are evaluated. The information obtained is used to develop the control room operator training in risk situations. Optionally, those solutions proven most effective are applied and tested in outage situations.

The work in this WP increases understanding on the level of stress and workload in simulated incident/accident situations and their effects on operator performance. This information can be used for developing the training as well as stress management protocols for exceptional operational situations.

The understanding of the positive and negative effects of stress is shared with the plant personnel throughout the project via annual workshops and practical demonstrations. The information will be shared in several stages, first as general information concerning the physiology and the stress response, and the stress management. These
will form a basis for development of simulation training. The results obtained in this work package are transmitted to the power plant community in workshops, as training material, as well as in the form of research reports.

2.4.4 Specific research questions of WP4

- What is the level of operator stress in simulated accident scenarios?
- What is the impact of stress on operator performance in simulated accident scenarios?
- What is the relationship between physiological and self-reported stress indicators?
- What is the impact of specific events within the simulated incidents and accidents to operator stress and recovery?
- How does stress affect the operator’s information seeking, information processing and problem solving ability?
- Are there individual differences in operators’ response to and recovery from stress?
- How does the role of the operator moderate the stress response?
- How do the experiences and results obtained in the simulated environments relate to the actual events of the Daichii Fukushima accident?
- How to help operators and simulator instructors to better understand the role of stress in operator work as well as in training, and how to help them to reduce the harmful effects of stress?

2.4.5 Specific results of WP4

- An objective, continuous, and non-invasive method for measuring operator stress in demanding situations
- An objective non-invasive method for evaluating NPP operator crew performance in demanding situations
- New knowledge about the level of stress in operator work during different operator work tasks
- Better understanding on how the physiological stress response influences operator performance
- Better understanding of prospects and limitations of self-assessment tools and the objective measurements of stress in complex environments, such as in NPPs
- Better understanding of how stress should be managed in both operator work and taken into account in operator training
- Training material and/or course on the significance of stress on operator performance
- Training material and/or course on the means of stress management on operator performance
- Training material and/or course on simulator instructor training on consequences of stress

Partner and person months allocated to WP4 to be given in the table.

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2.4.6 Task1 (T4.1) Objective measurement of acute stress response in simulated risk situations

2.4.6.1 Progress in 2015-16

During the first half of 2015, we tested various measurement devices and protocols in order to determine the most suitable measurement setups for NPP simulation settings. The devices were evaluated in terms of data quality, appropriateness and non-intrusiveness. In addition, we examined the relationship between the physiologically measured (heart rate variability, HRV and electrodermal activity, EDA), and self-evaluated (NASA task-load index, TLX) stress and mental load during demanding work simulation tasks in the laboratory, and on field.

The laboratory measurements combining HRV and EDA provided the highest accuracy in predicting stress and performance. The field measurements were confirmed feasible in evaluating the load of the participants, even though the accuracy as well as specificity were somewhat lower than in the laboratory. The Moodyring proved to measure the EDA activity accurately, but due to issues in reliability of raw data storage, these devices were replaced
with Empatica E4 wristworn EDA sensors in NPP measurements. As a new generation of Faros180 and Faros 360 sensors have proven superior to the Firstbeat devices in HRV data quality, feasibility, reliability and comfort of use, those were selected for the NPP measurements.


During the second half of 2015, our aim was to measure and quantify the stress and workload of the NPP control room operators in incident and accident situations, and evaluate the effects of stress and workload on performance. The power plant personnel, e.g., simulation trainers were interviewed and consulted. Different types of incident and accident situations were identified varying in their expected cognitive demands, and their perceived level of threat/ outcome health threat. In a pilot experiment, the best-proven technology (Faros 180/360 and Empatica E4) was used to collect ECG, HRV, EDA and accelerometer data from one test crew of four persons. The pilot experiment included three different simulated incident/accident scenarios varying in the degree of task difficulty (i.e., cognitive demands) and incident severity, selected on the basis of previous work, and trainer interviews. Preliminary analysis of the pilot data revealed a physiological stress pattern very close to theoretical predictions of our preliminary task modelling, and also close to the stress predicted by the simulator trainer. Both the EDA and the heart rate increased during the incident and accident scenarios and decreased during normal operation and after the simulation. The physiological stress was the highest in the scenario with the most severe health outcome, and lowest in the more routine testing scenario. The pilot data suggested also that the role of the operator may affect the stress reaction: the shift supervisor may be the most stressed in general, but the turbine operator showed the strongest physical stress response both in the more routine scenario requiring mainly the turbine operator’s control actions, and in the scenario with the most severe health outcome. The data quality was evaluated as high, with less than 15% artefacts, and the measurement devices functioned reliably throughout the recording.

The actual measurements of 6 operator crews with 4 operators each (N=24) were carried out during November and December 2015 at the TVO Olkiluoto NPP.

During the first half of 2016 the first level data analysis on physiological and experienced stress as well as predicted load during the simulations was carried out. In addition, 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. Two conference papers were prepared and presented by Pakarinen and Torniainen in EHPG2016 conference in May: Pakarinen S, Korpela J, Torniainen J, Laarni J, Karvonen H. “Control Room Operator Stress and Cardiac Activity in Simulated Incident and Accident Situations”, Proceedings of the 39th Enlarged Halden Research Project Meeting, Fornebu, Norway, May 9th – 12th, 2016. Torniainen J, Korpela J, Pakarinen S. “Control Room Operator Stress and Electrodermal Activity in Simulated Incident and Accident Situations”, Proceedings of the 39th Enlarged Halden Research Project Meeting, Fornebu, Norway, May 9th – 12th, 2016. In addition, a compilation of the results were presented by Pakarinen in the International Organization of Psychophysiology (IOP) Annual Meeting in Havana, Cuba and published in the Proceedings of the 18th World Congress of Psychophysiology (IOP2016), International Journal of Psychophysiology, 2016 Oct;108:73-74 http://dx.doi.org/10.1016/j.ijpsycho.2016.07.238. These two studies indicated that the power plant operators’ physiological as well as experienced stress levels significantly increased during the simulated scenarios as compared with the baseline. The scenario simulating a fire in the NPP was the most stressful. In addition, the analysis showed that the effects are not attributable to physical activity as the physical activity explained only small part of the variance in the stress measures for heart rate and subjective stress estimates, and almost no variance in the heart rate variability and electrodermal activity data. NPP operators varied in their sensitivity to perceive and or report their stress levels.

During the second half of the 2016, an analysis identifying the specific significant events that occur in all six simulation runs was conducted by Hannu Karvonen (WP3, VTT) together with the expertise from operator instructors of TVO Olkiluoto. On the basis of this work, Jussi Korpela (WP4, FIOH) calculated the durations of these events in order to quantify the response times of the crews. These analyses will be continued during 2017, with the goal of complementing the analysis with task-related physiology measures (WP4) as well as more detailed qualitative performance modeling (WP3: Supporting operational resilience in complex and dynamic environments).
Three full-page newspaper articles have also appeared on the topics of operator stress and decision making, featuring the research carried out in WP4 and partially also on WP3, in the following media: Tekniikka ja Talous 28.8. (http://www.teknikkalavalta.fi/paivain_lehti/testeijä-vydnimoimatalayontekijoilla-stressissä-sydän-sykki-kujin-metronomi-6577406), Talouselämä 1.9. (http://www.talouselama.fi/tyoelama/stressi-vaikkeutta-paatoksentekoa-6578560), Metalliteknikka 9/2016, p 28.

2.4.6.2 Year 2017

**Goals.** The data collected at NPP during 2015 will be re-analysed in order to quantify the amount of stress and workload of operators during specific events within the simulated accident and incident scenarios, and to evaluate the effect of stress on operator crew performance. This analysis may also provide new insights on stepwise recovery from stress from the point in which the solution of the problem is found, to when the situation has been fully resolved.

In addition, quantitative and qualitative measures (collaboration with WP3) that reflect operator and crew performance are identified and compared to the stress levels in order to evaluate the effects of stress on performance.

**Content.** The physiological data (ECG, EDA and movement data) collected in 2015 at TVO Olkiluoto simulator will be reanalysed with respect to specific events identified from the scenarios (e.g., fire alarms, fire is confirmed, electrical insulations executed, permission to extinguish the fire is given, gas leak detected). This will produce a set of significant events and the corresponding physiologically derived stress level at the time of and following the event. By quantifying stress in relation to distinct events within scenarios, it is possible to evaluate whether the events induce stress (e.g., fire alarm) and/or promote recovery from stress (e.g., electrical insulations executed, or temperature is returning to normal). The information from the movement sensors will be used to correct for the potential confounding effects of physical activity. Also crew performance will be evaluated e.g., as the time distance between certain events, such as fire confirmed and permission to extinguish the fire given, or gas leak detected and the source of leak identified.

**Measures.** Physiological signals (electrocardiography (ECG) and heart rate variability (HRV), electrodermal activity (EDA), and accelometry, self-evaluations, and trainer evaluations, performance measures (e.g., time taken to compete the task).

**Deliverables.** A peer-reviewed article on the operator stress during simulated incident and accident scenarios in a full scale simulator. In addition, an outline of a research article on the topic of stressfulness of distinct events in accident simulations and relationship between stress and performance.

2.4.6.3 Year 2018

Numerous studies have shown that task performance measurement is more complicated than merely the sum of individual operators’ performance, because teamwork relies heavily on communication, supervision of a common situation, and the sharing of the workload (Carvalho and Vidal, 2007; Sebok, 2000). The operators have been assigned roles, responsibility, and areas of specialization, which has two implications for stress management in the power plants: On one hand, the stress and workload of the individual is closely related the role of the individual. The shift supervisor often experiences higher stress and workload as compared with the other crew members, i.e., reactor operator and turbine operator. On the other hand, in ideal situations their partly overlapping training allows for effective management of stress and workload between the individuals by means of support and communication from other crew members, especially in challenging situations (Carvalho and Vidal, 2007; Sebok, 2000).

During 2018, the effect of operator role will be investigated from the data measured in the full scale simulator (Task 4.1, collection in year 2015; event related analysis in year 2017). This provides us with a two level analysis of i) stress in response to distinct events and crew performance in realistic (simulated) incident and accidents ii) detailed information on individual level stress and performance related to operator role. By combining the information, our aim is to evaluate the effect of a power plant operator’s role (reactor and turbine operator, shift supervisor) on stress and workload, as well as on individual and group performance in simulated accident and incident situations.

The data consists of measurements conducted on simulated accident and incident situations in Task 1 (T4.1) during 2015. The data will be analysed with appropriate advanced statistical analysis (most likely nonparametric test due to non-normal distributions and relatively small sample sizes) and computational methods. The group method of data handling (GMDH) algorithm will be applied to analyse group processes and predict group performance (see, e.g., Hwang et al., 2008).

At least one scientific publication in peer-reviewed journal series will be published by 2018.
2.4.7 Task 2 (T4.2) Management of stress in operator work and training

Due to repeated reductions in funding, Task 2 (T4.2) Stress management in operator work has been merged together with Task 3 (T4.3) Development of power plant risk scenario simulation training and renamed “Management of stress in operator work and training”.

2.4.7.1 Progress in 2015-16

Although stress is well known to affect performance, this information is not systematically taken into account in NPP personnel training. The first step for successful stress management is to make personnel aware of the direct practical consequences of the physical stress reaction to operator performance in accident and incident situations. This awareness has been increased by sharing information and educating the power plant operators throughout the study.

Operator training: In 2015, a lecture 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 was given on three occasions in TVO Olkiluoto NPP operator room personnel training. Focus was specifically on the significance of the stress management for performance in general and in NPP control room environment.

In 2016, the 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, evaluation of crew performance and possible effects of stress on performance on the basis of literature. The lecture was given by Pakarinen on Oct 3rd, and 11th and Oct 24th. In addition, NPP operators at Lovisa Fortum facilities were educated with a lecture on Stress and work load in NPP operator work presenting information on the effects of stress on operator performance and stress management on Oct 26th, Nov 3rd and Nov 24th.

Simulator training: In 2015, operator instructors were interviewed at the TVO Olkiluoto NPP and different types of scenarios were identified, varying in their degree of cognitive demands and severity of consequences. Three accident and incident scenarios were selected and the operator trainer evaluated the expected stress and load during these scenarios. These evaluations were compared to the operator crew members’ perception of stress and load, as well as to the physiologically quantified stress. According to our analysis carried out in 2016, the operator instructor’s evaluation of the stress and load for different types of the simulated scenarios closely corresponded to the physiologically derived stress of the operator crew, especially in the scenarios including a severe health outcome.

During 2016, our efforts focused on integrating the information gained from the simulator study to existing knowledge on operator training, and on distribution and dissemination of the findings. In spring 2016, Pakarinen, Torniainen, Karvonen and Laarni visited TVO Olkiluoto to present the current findings to the operator instructors and supervising personnel.

2.4.7.2 Year 2017

Goals. The aim of this task is to further increase operators’ simulator trainers’ awareness of the physiological stress response to the specific events included in the incident and accident scenarios as well as on the performance during these events and in real life situations such as the Fukushima accident.

Content. On the basis of current literature concerning management of stress and regulation of physiological stress responses, lessons learned from the Fukushima accident on decision making under pressure, as well as on the basis of results obtained in Task 1 (T4.1) training will be prepared. The power plant personnel will be given detailed information on the stress levels during the specific events of the incident and accident simulations and their effects on performance. In addition, they will be educated on the stress-related factors in the Fukushima accident. The goal is to deepen their understanding of the effects of stress in general but also in their own work at the NPP, as well as in actual accident situations. NPP crews are further instructed on the adaptive role of the acute stress response, leading to a positive cognitive appraisal and a more favourable physiological stress response. This approach will most likely be beneficial in situations of intermediate stress levels. The lessons learned from Fukushima will prepare for and increase understanding on operating under extreme stress. Furthermore, the personnel will be educated on practical methods to support decision making under pressure. One of the key aspects is in maintaining functionality and correctly timed responses (not too hasty reactions nor freezing) under extreme stress.

Measures. The results on stress, cognition and performance obtained from T4.1. combined with current literature on stress, cognition and performance, stress management and reports from Fukushima accident.
Deliverables. As a result of this task, a workshop presentation (slide set) for power plant personnel will be prepared by the end of 2017: Understanding the role of specific tasks, and operator role on stress and workload in simulated and actual power plant environments – Findings from simulated operator tasks and Fukushima accident.

2.4.7.3 Year 2018

The education on 2018 focuses on distributing the results obtained on Task1 (T4.1.) on the effects of operator role on the stress levels as well as performance on the simulation. Main emphasis is on bringing together the information on i) which aspects of the simulations are most stressful and which most recoverying ii) individual and role-related differences in reactivity to stress in these incident and accident scenarios. By identifying those events that are highly stressful but where the work load is not evenly distributed between crew members, measures could be taken to divide the load more evenly or add support to the most critical operations within the scenarios.

In 2018, a report on the distribution of stress between crewmembers in different scenarios will be prepared and the measures that might be necessary and/or possibly to ensure optimal performance will be jointly discussed with NPP personnel, especially operator instructors and supervisors.

2.5 Work package 5 (WP5) Supporting resilience in emergency management

2.5.1 Topic 1. Collaborative resilience in emergency exercises

According to Finnish regulations (716/2013; YVL C.5), nuclear power utilities should organize emergency exercises on a regular basis during the power plant’s operation. YVL C.5 states that “the objective of these exercises is to ascertain the appropriateness of the rooms, devices and equipment reserved for emergency response; the suitability, compatibility and scope of operating instructions and computer programs; and the operational capability of the organisation so that potential modification and improvement needs can be identified”. Similarly, international nuclear emergency exercises have been executed under OECD/NEA’s programme in the area of nuclear emergency management (INEX). Detailed guidance on the development and management of emergency exercises is provided, e.g., by IAEA’s report “Preparation, conduct and evaluation of exercises to test preparedness for a nuclear or radiological emergency” (IAEA, 2005). But more research is still needed in this area: For example, in the update to the Framework plan for 2017 it was stated that the operation of the emergency preparedness organisation and co-operation and interactions between different organisations should be further emphasized.

In emergency situations there is a requirement for the rapid mobilization of a dynamic inter-organizational system ranging from individual through organizational to system level (Comfort et al., 2010). According to Comfort et al., “the performance of the entire emergency management system depends on the iterative functions of scanning the environment for risk, detecting it accurately, verifying the degree of risk, analysing the information from the perspective of the whole system and transmitting the results in a timely manner to the multiple actors to serve as a basis for coordinated activity”. According to them, the emergency response centre can be seen as a bowtie that collects information from various sources, integrates, analyse and interpret the data from the perspective of the whole systems and then distribute the information to the relevant stakeholders that use the information to tune their operations (Comfort et al., 2010). From the perspective of resilience, it is important that it can be mobilized additional resources to the management of the emergency situation as the emergency situation reaches its culmination (Woods & Bran-lat, 2011).

In terms of resilience of emergency response systems, key social processes are collaboration, coordination, cooperation and communication; and efficient collaboration is not possible unless there are interrelationships between communication, cooperation and coordination (Saoual et al., 2014). Coordination and collaboration have been modelled using social network analysis (for a review, see, e.g., Durugbo et al., 2011). Different types of approaches have also been developed for the analysis of communication and information flow for organizations (for a review, see, e.g., Durugbo et al., 2013). In the present research, EAST (Event Analysis of Systemic Teamwork) method (Stanton et al, 2005) will be used. It has been developed to investigate the work of distributed teams in complex socio-technical systems. According to Walker et al., 2006), its approach integrates hierarchical task analysis, coordination demand analysis, communication usage diagrams, social network analysis, and the critical decision methods. Accordingly, the key descriptive constructs include who the agents are in a scenario, when tasks occur, where agents are located, how agents collaborate and communicate, what information is used, and what knowledge is shared.
In WP5, emergency responses and the related safety are approached from the perspective of emergency exercises the nuclear plants conduct with their collaborative partners. The EAST method is used to model the task, information and social networks in emergency exercises. It can also be useful in assessing both the resilience markers and potential weaknesses and failure points in the emergency response process (Stanton, 2013).

The focus of the research is in the nuclear power plant perspective as an organisation participating in emergency preparedness activities and especially in emergency exercises. Especially, this topic elaborates the collaborative practices and needs in emergency exercises.

2.5.2 Specific aims of WP5

The task aims to understand how emergency exercises are planned and conducted, and how these exercises are used to create routines for collaboration and coordination of activities. The EAST network modelling method is used in the analysis of task, social and information networks in distributed cognition in an emergency response system. The results will be used in the identification of resilience markers and possible weaknesses in the emergency response process and in the development of guidance for the improvement of the processes.

2.5.3 Specific research questions of WP5

- How emergency exercises are planned and conducted, and how these exercises are used to create routines for collaboration and coordination activities
- How different stakeholders collaborate and coordinate their actions in emergency exercises?
- How information is distributed and processed during emergency exercises?
- How to support communication and collaboration among exercise participants, regarding especially the Technical Support Centre and control room?

2.5.4 Specific research results of WP5

- Better understanding of challenges in planning and execution of emergency exercises
- New knowledge of communication and collaboration patterns in emergency response management
- Models of task, social and information networks in emergency exercises
- Better understanding of resilience markers and failure points in emergency response management.

Partners and person months allocated to WP5 to be given in the table.

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2.5.5 Task 1 (T5.1) Collaborative resilience in emergency exercises

2.5.5.1 Progress in 2015-16

During 2015, the conceptions of the key players participating in emergency exercises were studied, that is, those responsible for planning and organising the exercises in the two Finnish operating NPPs and one person responsible for participating and evaluating the exercises from STUK. Some exemplary reports of the exercises from both plants are also available for the study. From the theoretical point of view, emergency exercises were contemplated from the perspective of resilience engineering. From the more practical point of view, the structure, logics and practices regarding the emergency exercises as well as developmental needs, especially regarding collaboration, were revealed (reported by the end of January 2016).

During 2016, 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 spring 2016, the method for observing emergency exercises was developed. Due
to reductions in funding and because in autumn it was not possible to observe emergency exercise in neither of the operating plants, no emergency exercise observation was conducted. Instead, a part of the new methods was evaluated by some participants of the emergency exercises.

2.5.5.2 Year 2017

Goal. The aim is to observe and characterize the underlying structure, logics and practices of collaboration, coordination and communication during emergency exercises and chart the needs for the development of the emergency exercises in the nuclear domain from this perspective. The practices of Technical Support Centre will be focused on. The study is performed in collaboration with IFE; we will provide IFE information about the role and practices of the coordinator of the TSC in Finland. IFE gathers this type of information also from other sources and utilizes this information for the identification of the factors which seem important from the development of coordinator work point of view.

Content. Data collection over observations in an emergency exercise will be executed in 2017. The proceeding of the exercise will be observed from the perspective of the TSC so that in the research, TSC and its collaborative and coordinative parties are focused on from the resilience and collaboration perspectives. The observational data to be collected is to be analysed and interpreted, especially for modelling purposes, and based on observations and the related information, the practices and developmental needs of the TSC will be analysed, discussed and reported.

Methods. Event Analysis of Systemic Teamwork (EAST) method is used to analyse and model the activities of the Technical Support Centre in emergency exercises. Especially, the proceeding of the exercises in the plant and the TSC from the resilience perspective are focused on. The practical methods to be used are interview, structured observation and questionnaire.

Deliverable. Research report on collaborative resilience in emergency field exercises (by the end of 2017). Also a conference paper (EHPG) will be prepared.

2.5.5.3 Year 2018

In 2018, the work performed during years 2015-2017 is utilized to create an overview of the developmental needs and challenges and guidance to overcome these needs and challenges. Depending on the amount and quality of gathered information as well as the amount of funding, additional data may be collected. As a result, guidance is developed for the improvement of emergency response process in the nuclear domain, especially from the Technical Support Centre point of view.

2.6 Work package 6 (WP6) Applying a HF tool to learn to analyse human contribution to nuclear safety (HUMTOOL)

The mastery of Human Factors is known to be a necessary area of expertise, while human contribution to the safety incidents and accidents has been recognized already for a long time (Reason, 1997; Dekker, 2002; Reason, 2008). Still, technological and procedural contribution to safety have been dominated, and the human has not been in focus (Hollnagel, 2009; Kirwan, 2003; Teperi, 2012). The system safety models and frameworks have been changing during the decades, from technical analysis to human factors, safety culture and system analysis. (Hollnagel, 2006; Hale & Hovden, 1998; Reimann & Oedewald, 2009), i.e., self-reflection by learning to analyse variability of human action during operational experiences.

Nuclear safety guidelines recommend incorporating systematic methods in the management system in order to identify and manage human and organisational factors affecting safety. According to the guidelines, personnel’s individual competence should be developed in the identification and management of human factors and potential
errors. The strengths, weaknesses, and areas for improvement of the safety culture could be best identified in connection with operational events. (YVL A3, 303, 311, 319-320, 315)

The meaning of a reflective way of action and applying self-reflection skills at work and learning culture has been highly recognized in safety research and in safety standards of nuclear industry (e.g., Klein, 1998; IAEA 2006). YVL guide states it as follows: “The management and all organisational levels shall carry out self-assessment in order to evaluate and improve the performance and the safety culture. Self-assessment means that the organisation's personnel evaluate their own work performances or processes related to their work against pre-defined criteria” (YVL A10, 709).

This kind of safety management procedure has been tested and found useful in the studies of improving the mastery of HF in safety critical field of air traffic management (Teperi, 2012; Teperi et al. 2015). The HF tool has been used by operative personnel (air traffic controllers and their chiefs) since 2008 in reporting incidents, to find out, which kind of issues in their own actions or behavior (i.e. human factors) weakened or improved the safety of the case. The successes and failures of every-day-work have been in focus.

When operative personnel accepted the use of a HF tool several years ago (and the HF is still used by the operative personnel), and analysed the positive and negative causal factors of the incidents at their work, they learned to analyse the background factors of the incidents, and better understood the human contribution in occurrences and incidents emerging in their own work. The benefits of the HF tool were its visual appearance, user-friendliness and the congruence of its contents with existing HF tools such as HERA-Janus method and HFCAS, which had been used in aviation earlier. (Teperi et al., 2015).

It is also recommended at the safety critical areas, that safety-significant operational events have to be investigated and the further steps taken, in order to define corrective and preventive actions. Lessons learnt from the operational events (operating experience feedback) is crucial in order to prevent accidents and other events in future work (YVL A10; 101, 102; Dekker, 2007). However, practical, user-friendly tools may be missing. Most of the investigation or analysis techniques are designed for larger accident or case investigations, and they are proposed to be used by investigation experts (e.g., AcciMap or BowTie-method used by Accident Investigation Board of Finland; FRAM presented in 2009 by Leonhardt et al.). Furthermore, they are time consuming and need strong input in training (Teperi et al, 2015). In any case, as operative personnel are not professionals of human contribution, they would benefit from usage of a tool, with which they could improve their understanding and identifying of the HF contributing at they work and in operational events. They would also learn to analyse them straight after the operative event, as a part of their everyday work.

The HF tool consists of several items at four levels (individual-, work-, group- and organizational factors), which all have been stated relevant as background factors of mishaps in safety critical work. For this project, HF tool was modified to better suit the conceptions and work features of nuclear domain (see Figure 5).

During 2015-2016 research, a new study focus was found regarding the need for a peer-based mental first aid model (MFA) for the nuclear actors. This will be under evaluation, because of the fact that defence-in-depth also include mitigation and consequence management after the unwanted and unexpected events, especially if they are not fully severe, but still harmful and weakening the competence and capacity of the actors. Nowadays, these kinds of tools are missing in the safety management procedures of nuclear industry (despite of occupational health services in severe accidents), even though they are already in use in several other safety critical domains. This kind of model would be appropriate today, for example, because of the change management challenges going on at the NPPs in Finland, which processes cause several kind of reactions (possibly weakening work motivation and at worst, commitment to safety) among the personnel in the nuclear industry organisations. At least, the MFA model could be tested and the usefulness evaluated by some actors (several interviewees in STUK, Fortum and TVO revealed interest to the model) during 2017-2018. In the MFA model, SAFER-discussion frame based on international crisis psychology theory is used (Mitchell, 2006; Leonhardt & Vogt, 2006), to normalize the reactions and supporting recovery right after the (less severe, but) unwanted events. A similar kind of model has been piloted by five organisations in City of Helsinki in 2014-15 (Teperi et al., 2015b).

Furthermore, based on 2015-2016 findings, the field of HF was considered quite abstract, and there was a clear need to concretize the conception of HF, as well as to improve knowledge of HF, i.e., what does it mean in everyday life of the organization in individual-, work-, group- and at the organisational level. For example, research participants (both at the regulator and NPP side) recognized the need to improve the capability of summarizing the most evident findings revealed via Safety Management System (SMS) practices, as well as the activities in decision

---

1 Operational events, OE; such developments, failures, flaws and problems that are of relevance in terms of nuclear or radiation safety. (YVL A10, 104)
making, implementation and putting actions in practice, which should emerge as an agile process after identifying the development needs and recommendations in reporting and event analysis. These topics are in focus of the overall work of WP in 2017-18.

2.6.1 Specific aims of WP6

The WP6 aims to study appropriateness and use of the HF tool as an investigation method in operative event analysis of nuclear industry. Furthermore, we evaluate the implementation process of the HF tool - its effects on safety thinking and practices, as well as supporting and hindering factors of the implementation process. In the study, also currently used tools and models to analyse human contribution as a part of safety management are evaluated.

2.6.2 Specific research questions of WP6

- Do the key persons of the NPPs (HF and safety experts, supervisors, other personnel) concern the use of the HF tool useful; does it offer some added value compared to the currently used safety management practices of the organisations (NPPs, regulator)?
- Does application of the HF tool have effects on safety thinking or safety management procedures of the nuclear safety organisations?
- What are the supporting and hindering factors of the HF tool implementation process?

2.6.3 Specific research results of WP6

- Improved skills of recognizing and identifying human contribution in the safety critical work processes of nuclear safety organisations

Figure 5. HF tool being modified in nuclear industry in this project.
- Modelling of the currently used models and techniques used in data collection and analysis concerning human contribution in safety management by the companies; comparison to the usefulness of the new HF tool
- Modification and implementation of the HF tool to suitable parts of the current safety management system, i.e., reporting, risk analysis, investigation, training/coaching and/or work development activities
- Comparison of the HF tool usage to other and earlier applied HF related tools.

Partners and person months allocated to WP6 to be given in the table.

<table>
<thead>
<tr>
<th>Partners in WP6</th>
<th>Person months in 2016</th>
</tr>
</thead>
<tbody>
<tr>
<td>FIOH</td>
<td>3.5</td>
</tr>
</tbody>
</table>

2.6.4 Task 1 (T6.1) HF evaluation

2.6.4.1 Progress in 2015-16

In 2015, we identified the current HF conceptions, practices and tools used by the nuclear safety organisations (NPPs and STUK), as well as evaluated user experiences of the use of the HF tool in NPPs. In 2016, we started to validate and test HF tool, and recognized HF needs at NPPs. During 2016, we analysed the data collected in 2015 (interviews at Fortum, n=12, at TVO, n=8 and at STUK, a group interview of 2 participants) and safety documentation analysis regarding current HF guidelines and current practices (e.g. YVL guide, guidelines at each NPP). Also intervention material from two workshops (workshop I in April 2015 and workshop II in September 2015) was analysed accordingly (for details concerning workshops, see 2.6.5.). Results based on the data collected and analysed were reported in a manuscript of a scientific article (submitted 30th April, revised 25th September–25th November), as well as in Safety2016 conference in an oral and a poster presentation.

2.6.4.2 Year 2017

**Goal.** The aim is to plan and conduct HF tool implementations and developments based on recognized HF needs of the organisations.

**Content.** Work with the NPPs which are already participating will be continued, to evaluate the successfulness of the implementation process of the HF tool. We will also further assess the HF needs and disseminate the current results to STUK and Fennovoima. Some more research data (project material) will be collected and analysed in 2017.

**Methods.** Three group discussions/interviews with 1) already participating NPPs, 2) Fennovoima, and 3) STUK; data collected in 2015-16, and analysed in 2016 (HF-needs, HF tool user experiences) will be utilized.

**Deliverable.** Article manuscript, regarding HF related needs in nuclear industry, especially from the macro-ergonomics and system thinking point of view.

2.6.4.3 Year 2018

In 2018, the focus of HF evaluation is to evaluate HF implementation process, its hindrances and success factors at the whole nuclear industry domain. Main findings concerning HF needs at the nuclear industry (NPPs, authority, project Fennovoima), as well as needs, possibilities and resources to implement the HF tool and user experiences regarding HF tool will be revealed, and based on those findings, recommendations for future HF actions will be suggested. Co-operation at system level, with different nuclear actors, will be emphasized. The scope is 1) to form an overall picture of the level and development phase of the HF implementation in nuclear industry in Finland (and possibly, internationally); 2) to investigate what the basic competence needs and corrective actions would be useful at the field of nuclear industry regarding HF; and 3) to design ‘a HF toolkit’, based on scientifically sound findings, for the industry actors from the regulator side to an operative use, as well as to the training partner of the nuclear
industry (e.g., LUT). If opportunities emerge to test or coach the MFA model (during 2017) and if it has proven to be useful in nuclear domain, that will be included to the ‘HF toolkit’.

2.6.5 Task 2 (T6.2) HF tool workshop

2.6.5.1 Progress in 2015-16

In 2015, two workshops with the NPPs were conducted: one-day-workshop I with two HF-experts of NPPs and two-day workshop II with eight safety experts from TVO (4) and from Fortum (4), in which three operational events were analysed with both the currently used HF method (AcciMap modification) and with the newly offered HF tool. In 2016, research data from the workshops, interviews, and documentation analysis was analysed. 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. HF tool was regarded as easy-to-use, and it was considered a useful tool especially at OE analysis, reporting and training. Data analysis revealed needs to concretize the conception of HF, and to broaden the use of the HF conception to include macro-ergonomics and system thinking. In autumn 2016, the NPPs have independently analysed some of their OEs with the HF tool. Workshop III is arranged in November 2016, where the analyses conducted by the NPPs are discussed and more experiences of the HF tool use will be collected (results not available at the moment of the funding application process).

2.6.5.2 Year 2017

**Goal.** The aim is to further identify and test the different applications of using the HF tool and HF thinking in different nuclear power organisations and environments; in operating NPPs as well as in NPP projects and with the authority, to broaden the view of HF to a more system-wide scope.

**Content.** We will further identify the possibilities to use HF tool and disseminate HF thinking with several nuclear actors: 1) already participating NPPs, 2) Fennovoima project, and 3) STUK. The HF tool is also compared to the processes and outputs of other investigation methods. The implementation process of the HF tool in the NPPs will be evaluated. Validation of the HF tool is continued by analyzing user experiences collected from the NPPs in October-November 2016, while using HF tool in analysing few operational events in the NPPs. The nuclear experts will be supported by the research group while further training the use of the HF tool.

**Methods.** Workshop IV with the safety experts of the currently participating NPPs will be conducted, to continue the validation and supportive process of the use of the HF tool. Also, a workshop with Fennovoima will be conducted, to assess the applicability of HF tool in nuclear power plant projects. If possible, a workshop with STUK will be arranged, to assess the applicability of HF tool for the authority.

**Deliverable.** Workshops with NPPs, STUK and Fennovoima.

2.6.5.3 Year 2018

Further actions in 2018 are planned in co-operation with the NPPs, Fennovoima project and STUK, depending on the client needs and resources as well as concerning the results and findings of the workshops in 2016-17. There is still a need to continue workshops for the same participants of TVO and Fortum (safety experts) that were available at 2015-16, to observe the process of utilizing the HF tool, and to find out the pros and cons of implementing the tool in the current safety management procedures. Also workshop results with new groups, in the NPPs, Fennovoima project and STUK will be analyzed. These workshops would create research data for further coaching and validation of the HF tool.

2.6.6 Task 3 (T6.3) Modifying HF tool

2.6.6.1 Progress in 2015-16

The translation of the HF tool to suit the context of nuclear safety was conducted successfully in 2015-2016. November 2016 workshop will offer more insight to the HF tool modification needs. The HF tool earlier presented
in Finnish was translated into English. The modified HF tool was published as a part of an article, and Safety2016 congress deliverables in Tampere in 9/2016.

2.6.6.2 Year 2017

Training material of the HF tool will be modified, to be more user-friendly for the users (e.g., Lectora software). Co-operation with SAFIR-MAPS project (in September 2017) and CORE WP1 (in March-April 2017) will be arranged to further modify HF tool, e.g., to support inter-organizational view of safety management as well as a tool for perceiving success factors of the OEs.

2.6.6.3 Year 2018

In 2018, the aim will be to modify and implement the HF tool to suitable parts of the current safety management system, i.e., reporting, risk analysis, investigation, training/coaching and/or work development activities. Also, a more proper comparison of the HF tool usage to other and earlier applied HF-related tools in nuclear domain is conducted. A digital version of the HF tool will be produced. The action would be a part of other FIOH digitalization progress activities.
### 3. Deliverables and milestones 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable and milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Presentation and workshop to be held at HUSC meeting in 7.-8.11.2017</td>
<td>0.5</td>
<td>8.11.2017</td>
</tr>
<tr>
<td>D1.2.1</td>
<td>Scientific publication (conference article for the REA Symposium) (selected as a milestone)</td>
<td>2.5</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Conference article (for the REA Symposium) on the development of training in NPPs jointly with 2.3 (selected as a milestone)</td>
<td>2.7</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D2.3.1</td>
<td>Conference article on the development of training in NPP jointly with 2.1</td>
<td>N/A</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Preliminary interviews conducted; the draft version of the questionnaire completed (selected as a milestone)</td>
<td>1.0</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Slide set on multitasking modelling</td>
<td>0.4</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.2.1</td>
<td>Conference article on the modelling of troubleshooting performance</td>
<td>0.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.2.2</td>
<td>Conference article on cognitive heuristics in operator work</td>
<td>0.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D4.1.1</td>
<td>A peer-reviewed article on the operator stress during simulated incident and accident scenarios in a full scale simulator.</td>
<td>2.5</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D4.1.2</td>
<td>An outline of a research article on the topic of stressfulness of distinct events in accident simulations and relationship between stress and performance.</td>
<td>1.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D4.2.1</td>
<td>A workshop presentation: Understanding the role of specific tasks, and operator role on stress and workload in simulated and actual power plant environments – Findings from simulated operator tasks and Fukushima accident..</td>
<td>1.0</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>D5.1.1</td>
<td>EAST-based communication analysis method developed (selected as a milestone)</td>
<td>0.5</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D5.1.1</td>
<td>Research report on collaborative resilience in emergency field exercises</td>
<td>1,5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D6.1.1</td>
<td>Article draft based on 2015-2016 data collection: HF needs and possibilities at nuclear industry/system level (selected as a milestone)</td>
<td>1,0</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D6.2.1</td>
<td>Research report of the workshops with NPPs and STUK; implementation and development processes and outputs</td>
<td>1,5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D6.3.1.</td>
<td>Workshops with MAPS and CORE WP1, e.g. to formulate new views to HF tool</td>
<td>1,0</td>
<td>31.12.2017</td>
</tr>
<tr>
<td><strong>Total pm</strong></td>
<td></td>
<td><strong>18.6</strong></td>
<td></td>
</tr>
</tbody>
</table>
4. Project organisation

VTT is fundamentally responsible for the project.

Project’s persons in charge are the following:

- **Project manager**: Jari Laarni (VTT)
- **Deputy project manager**: Satu Pakarinen (FIOH)
- **Project secretary**: Saija Valkama (VTT)

Persons in charge in each work package:

- Work package 1. Kaupo Viitanen (VTT)
- Work package 2. Mikael Wahlström (VTT)
- Work package 3. Jari Laarni (VTT)
- Work package 4. Satu Pakarinen (FIOH)
- Work package 5. Marja Liinasuo (VTT)
- Work package 6. Anna-Maria Teperi (FIOH)

Halden In-kind funding is allocated to WP5.

*Table 4. Project personnel in 2017 in alphabetical order per company.*

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hannu Karvonen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T3.1 - T3.2</td>
<td>0,5</td>
</tr>
<tr>
<td>Hanna Koskinen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1; T1.3; T5.1</td>
<td>1,5</td>
</tr>
<tr>
<td>Timo Kuula</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T2.1 - T2.3</td>
<td>0,5</td>
</tr>
<tr>
<td>Jari Laarni</td>
<td>Principal scientist</td>
<td>VTT</td>
<td>T3.1 – T3.2</td>
<td>1,9</td>
</tr>
<tr>
<td>Marja Liinasuo</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T1.1; T1.3; T5.1</td>
<td>1,5</td>
</tr>
<tr>
<td>Markus Porthin</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T5.1</td>
<td>0,5</td>
</tr>
<tr>
<td>Kaupo Viitanen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1; T1.3</td>
<td>1,5</td>
</tr>
<tr>
<td>Mikael Wahlström</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T2.1 – T2.3; T3.3</td>
<td>1,5</td>
</tr>
<tr>
<td>Heli Heikkilä</td>
<td>Researcher</td>
<td>FIOH</td>
<td>T2.3</td>
<td>0,1</td>
</tr>
<tr>
<td>Jussi Korpela</td>
<td>Researcher</td>
<td>FIOH</td>
<td>T4.1; T4.3</td>
<td>1,0</td>
</tr>
<tr>
<td>Kristian Lukander</td>
<td>Researcher</td>
<td>FIOH</td>
<td>T4.1; T4.3</td>
<td>2,0</td>
</tr>
<tr>
<td>Satu Pakarinen</td>
<td>Specialized researcher</td>
<td>FIOH</td>
<td>T4.1; T4.3</td>
<td>2,0</td>
</tr>
<tr>
<td>Vuokko Puro</td>
<td>Research engineer</td>
<td>FIOH</td>
<td>T6.1 – T6.3</td>
<td>1,2</td>
</tr>
<tr>
<td>Henriikka Ratilainen</td>
<td>Research engineer</td>
<td>FIOH</td>
<td>T6.1 – T6.3</td>
<td>0,55</td>
</tr>
<tr>
<td>Marika Schaupp</td>
<td>Researcher</td>
<td>FIOH</td>
<td>T2.3</td>
<td>0,5</td>
</tr>
<tr>
<td>Laura Seppänen</td>
<td>Researcher</td>
<td>FIOH</td>
<td>T2.3</td>
<td>0,1</td>
</tr>
<tr>
<td>Anna-Maria Teperi</td>
<td>Specialized researcher</td>
<td>FIOH</td>
<td>T6.1 – T6.3</td>
<td>1,2</td>
</tr>
<tr>
<td>Maria Tiikkaja</td>
<td>Researcher</td>
<td>FIOH</td>
<td>T6.1 - T6.3</td>
<td>0,55</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>18.6</strong></td>
</tr>
</tbody>
</table>
5. Risk management

Significant and intolerable risks are printed in red in table below. Risks management actions for these risks are presented in 5.2.

5.1 Identification and evaluation

<table>
<thead>
<tr>
<th>Instruction:</th>
<th>PROJECT RISK EVALUATION</th>
<th>Impact / consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td>A. Pick up relevant risks from the check list into the table below (see app.) or enter other identified risk.</td>
<td>PROBABLE (3)</td>
<td>LOW (1) Moderate risk</td>
</tr>
<tr>
<td>B. Name the risk and estimate its likelihood (1-3).</td>
<td></td>
<td>HARMFUL (2) Significant risk</td>
</tr>
<tr>
<td>C. Evaluate risk’s impact /consequences (1-3).</td>
<td>POSSIBLE (2)</td>
<td>SERIOUS (3) Intolerable risk</td>
</tr>
<tr>
<td>D. Choose RISK CLASS (1-5) according to this table.</td>
<td>UNLIKELY (1)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Likelihood</th>
<th>Project objectives, definitions, tasks, maturity of the relevant technology</th>
<th>Impact</th>
<th>Risk class</th>
</tr>
</thead>
<tbody>
<tr>
<td>PROBABLE (3)</td>
<td>External mandatory rules and regulations (customer, legislation, customs rules).</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>IPR rights/ restrictions.</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>POSSIBLE (2)</td>
<td>Project’s human resources</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>UNLIKELY (1)</td>
<td>Understanding and acceptance of the project objectives and research methodology (commitment of the project group).</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td>Project group’s ability to cooperate internally/ externally.</td>
<td>1</td>
<td>2</td>
</tr>
</tbody>
</table>

| | Timetable or cost pressures and financing | Impact | Risk class |
| | The project has challenging timetable and cost targets. | 2 | 3 | 4 |
| | Possibilities/ limitations to align timing (etc.) with other unfinished or planned projects. | 2 | 2 | 3 |
| | Vulnerability to impacts/ changes in external conditions (e.g. temporary dismissals). | 2 | 2 | 3 |

| | Project’s external stakeholders / cooperation with subcontractors | Impact | Risk class |
| | Match of the organization cultures. | 1 | 2 | 2 |

| | Equipment, premises and infrastructure (project/ research environment) | Impact | Risk class |
| | Possible equipment failures and poor equipment availability that may lead to interruptions or delays in the research work. | 1 | 3 | 3 |
| | Problems in accomplishment of the proposed research activities at Fortum Lovisa and TVO Oikiluoto plants. | 2 | 3 | 4 |

| | Occupational safety, environmental and information security risks | Impact | Risk class |
| | Information security risks (computers/ PCs, tablets, smartphones, telecommunication, inadequate security culture etc.). | 2 | 2 | 3 |
| | General employee safety risks (traveling, hazards related to geopolitical circumstances/ political unrest etc.). | 2 | 1 | 2 |
## 5.2 Risk management actions for significant and intolerable risks (risk class 4-5)

<table>
<thead>
<tr>
<th>Risk name:</th>
<th>The project has challenging timetable and cost targets.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Description and impact:</td>
<td>Ambitious goals of the project cannot be reached within the proposed timeframe and with the allocated resources.</td>
</tr>
<tr>
<td>Evaluation date:</td>
<td>Contingency plan and decisions</td>
</tr>
<tr>
<td></td>
<td>Regular follow-up of the progress of WPs and resource use.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Risk name:</th>
<th>Problems in accomplishment of the proposed research activities at Fortum Loviisa and TVO Olkiluoto plants.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Description and impact:</td>
<td>Simulator tests cannot be arranged in a planned time schedule and/or in a planned extent.</td>
</tr>
<tr>
<td>Evaluation date:</td>
<td>Contingency plan and decisions</td>
</tr>
<tr>
<td></td>
<td>Early discussion with contact persons on appointment issues.</td>
</tr>
<tr>
<td></td>
<td>Reduction of the extent of testing.</td>
</tr>
</tbody>
</table>
References


IAEA (2005). *Preparation, conduct and evaluation of exercises to test preparedness for a nuclear or radiological emergency*. Vienna: IAEA.


The Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2018)
Resource Plan for 2017

**CORE**
Crafting Operational Resilience in Nuclear Domain

### Expenses and Financing

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat&amp;supp</th>
<th>Travel</th>
<th>Ext serv</th>
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### Comments:
- Domestic travels (Loviisa); Resilience Engineering Symposium (Belgium 26.-29.6.2017); HUSC seminar in Sweden in 7.-8.11.2017: 3000€
- Resilience Engineering Association (REA) Symposium, Liège, Belgium, 26-29.6.2017: 1200€
- RG/ad hoc meeting service costs: 500€
- Domestic travels (Loviisa, Olkiluoto); travel to EHPG 2017: 2500€
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Comments:

- T2.1-T2.3  Travel: NPP visits: 500€; other costs: transcriptions etc. = 500 €
- T4.1  Open acces fee of conference proceedings/journal article and poster printing = 2000 €
- T4.2  Training workshops 3 visits to TVO and 3 to Fortum, travel costs, daily allowance 5000€/visit = 3000 €
- T6.1-6.3  Travel: Participating WOS2017 Safety management complexity in changing society (Czech Republic); 1 researcher = 2500€
- T6.1-6.3  Travel: NPP visits/workshops etc. = 2000€; other costs: materials, transcription etc. = 1000 €
SAFIR2018 Project plan

ESSI
Electric Systems and Safety in Finnish NPP

Seppo Hänninen, Riku Pasonen, Anna Kulmala
VTT Technical Research Centre of Finland Ltd

Matti Lehtonen
Aalto University, School of Electrical Engineering
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1. Research theme and motivation

After the Fukushima accident, the risk of common cause failures in NPP electric systems has come to focus. In 2016, a preliminary study was carried in SAFIR programme about the most important research needs with regard to the electric systems in Finnish NPP:s. This research proposal is based on this preliminary study.

1.1 Background and state-of-the-art

In the preliminary study, with regard overvoltages and disturbances in electrical systems, three important possible sources of common cause failures were identified:

Open phase condition (OPC). Open phase condition may happen if one or two phases of circuit breaker of the transmission grid do not close properly. The impact of OPC depends on the severity of unbalance. Generators are relatively sensitive to unbalance and they are easily tripped by unbalance relays. In directly connected motors, sustained operation with unbalanced supply voltage leads to either tripping of the motor protection, and hence disconnection of the motor from power supply, or in the worst case, damaging of the machine due to overheating. If unbalance is sustained for a longer time, transformers and cables may be overheated due to excessive load currents. In case of converters, the uninterruptible AC power system functionality can be impacted. The risks of OPC can be managed if there are means for timely warning of OPC situation, and if the NPP operation personnel is trained to take swift mitigating actions.

Lightning overvoltages can be caused in NPP power and automation systems via two different mechanisms: If lightning strikes in the phase of the 400 kV (or 110 kV) transmission line connected to the NPP, an overvoltage can be transmitted through the transformers to the lower voltage levels. The second mechanism is lightning strike to the grounded parts of the NPP, causing Ground Potential Rise (GPR) and potential differences between grounded parts and insulated power, communication and instrumentation circuits. The overvoltage protection is typically rated for 200 kA lightning strikes, which is considered to be the maximum expected lightning current. However, theoretically it is possible that this maximum current is exceeded, and it may be useful to examine the probability of higher lightning currents and the impact they may have to the NPP electrical and automation systems.

Adaptive operation of NPP is expected in future flexible power system due to fluctuating renewables power. A major energy system transition is underway in Europe. The European energy strategy, through its Directive 2009/28/CE sets ambitious goals for the energy systems of the future. The strategy implies a substantial increase of the share of renewable electricity production in the EU. This can only be achieved if solutions are found to keep the electricity system stable while having larger shares of renewable energy sources connected to the network at all voltage levels. This turns out to be a very difficult problem to solve. In presence of a considerable share of renewable sources of electricity which depends on weather conditions, this balance is more difficult to be achieved. Consequently, managing fluctuations in production and consumption is getting ever more challenging. Many European countries, Finland among them, will retain nuclear power as part of their future energy systems.
Nuclear generation is emissions-free, and current nuclear power plants can technically perform a wide range of production control, as is being routinely practiced in countries like Germany and France, where the share of renewables in the national mix is large. It is recognized that nuclear power will continue to be generated in Finland, and this creates a challenge of combining growing levels of small-scale distributed production with nuclear power. It is also possible in Finland that periodically the NPPs have to adapt to the consumption and run at decreased power level after all new NPPs are taken in use.

The adaptive operation of NPP has not been thought very much in Finland. From the point of view of the safety it is not recommended up driving/down driving and it is also a usability matter. It is safer to run NPP on a lower reactor effect. The failures in operation and heat transients can be affecting safety. It is advantageous from the point of view of the fuel when NPP is running on lower power level. The disturbance sensitivity of the electrical components and ICT systems increases, and it may also lead to problems in stabilization of the national grid. In that sense, potential safety issues of electrical systems related to the operation of nuclear power plants in the load following mode need research.

1.2 Objectives and expected results

The objectives of research are to examine the possible common cause fault impacts of OPC and large lightning strikes in Finnish NPP electrical systems. Also the risks of adaptive operation of NPP in load following mode will be examined. Analysis is made about following expected results:

OPC: severity of unbalance in different possible open phase situations is analysed. The consequences of possible voltage unbalances are assessed from different electrical system components point of view, and criticality of different OPC cases is analysed with regard the possible risks and time required from mitigation actions. An expected result will be how the operating plants have been prepared to OPC.

Lightning overvoltages: Voltage stresses caused by large lightning currents in Finnish NPP electric systems is analysed for both surges entering via transmission grid and through lightning strikes into the grounding system. The adequacy of overvoltage protection is assessed and possible improvements are suggested for the surge arrester sizing and location, as well as for the grounding arrangements of the electrical, automation and instrumentation systems.

Concerning the adaptive operation of NPP, the objective is to estimate requirements, technological limits and risks of adaptive control in today’s nuclear power plants with regard to electrical systems in order to avoid the increase of disturbances in power plant. The objective in 2017 is to obtain the minimum requirements for the manoeuvrability capabilities of Finnish nuclear power plants.

1.3 Exploitation of the results

The results of OPC related research will be exploited by developing early detection solutions for unbalance condition in the NPP electric systems. Another important issue is to provide NPP operation personnel understanding about the time criticality of the OPC situation and possible means of mitigating the situation by operation decisions.

The results of lightning research will be utilized in improving the lightning overvoltage protection and grounding arrangements of the NPP electric, automation and control systems, if deemed necessary.
The results of the research regarding adaptive operation of nuclear power plant can be exploited to setting the technological limits of adaptive control in today’s nuclear power plants with regard to electrical systems in order to avoid the increase of disturbances in power plant.

The Table 1 below gives examples of the main results, their intended users and the times when the results will be available.

Table 1. Main results of the ESSI project and their availability.

<table>
<thead>
<tr>
<th>Result</th>
<th>End user types</th>
<th>Project year</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Regulator</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Utility</td>
<td></td>
</tr>
<tr>
<td></td>
<td>TSO</td>
<td></td>
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<tr>
<td></td>
<td>Research</td>
<td>2017</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2018</td>
</tr>
<tr>
<td>Preparenedness of the operating plants afainst OPC</td>
<td>X X X X X X</td>
<td>X X</td>
</tr>
<tr>
<td>Impacts of unbalances caused by OPC on different electrical systems and components in the NPP</td>
<td>X X X X X X</td>
<td>X X</td>
</tr>
<tr>
<td>Methods for timely warning of OPC condition in NPP electric systems</td>
<td>X X X X X X</td>
<td>X</td>
</tr>
<tr>
<td>Recommendations for the NPP operation in order to mitigate OPC condition in Finnish NPP:s</td>
<td>X X X X X X</td>
<td>X</td>
</tr>
<tr>
<td>Analysis of GPR in Finnish NPP</td>
<td>X X X X X X</td>
<td>X</td>
</tr>
<tr>
<td>Recommendations for lightning overvoltage protection in NPPs in case of GPR</td>
<td>X X X X X X</td>
<td>X</td>
</tr>
<tr>
<td>Modelling and simulation of the lightning over voltages penetrating from the 400 kV or 110 kV grid to the NPP electric systems</td>
<td>X X X X X X</td>
<td>X X</td>
</tr>
<tr>
<td>Possible recommendations for the coordination of overvoltage protection measures for lightning surges coming from transmission grid. Optimal grading of surge arrester protection levels.</td>
<td>X X X X X X</td>
<td>X X</td>
</tr>
<tr>
<td>Modelling and simulation of LV AC and DC systems and overvoltage protection</td>
<td>X X X X X X</td>
<td>X X</td>
</tr>
<tr>
<td>Risks of adaptive control to NPP electrical systems and stability of the grid</td>
<td>X X X X X X</td>
<td>X X</td>
</tr>
<tr>
<td>The minimum requirements for the maneouvrability capabilities of Finnish nuclear power plants</td>
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<td>X X</td>
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</tbody>
</table>
1.4 Appropriateness of the project to SAFIR2018 programme

The objectives of ESSI project are related to SAFIR2018 research area of “Plant safety and systems engineering” and especially to the “SAFIR2018 – Update to the Framework plan for 2017 call” published by SAFIR2018 Management Board 18.8.16.

The project meets the goals set in the SAFIR2018 framework plan for the analysis of electric systems. The research topics selected are based on preliminary study performed in 2016. In this study several possible factors that may lead to common cause failures were identified. After the Fukushima accident, it became necessary to increasingly focus on the development of the reliability of the electrical distribution systems on NPPs. The probability of the occurrence of common cause failures and the severity of the consequences must be reduced from the perspective of the entire plant’s safety. In the Table 2, we describe how the ESSI project supports the goals of the programme as stated in the in the SAFIR2018 framework plan and in the Update to the Framework plan for 2017 call.

Table 2. The objectives of ESSI linked to the goals of the SAFIR2018 programme and its updated call for 2017.

<table>
<thead>
<tr>
<th>SAFIR2018 themes and goals</th>
<th>ESSI themes and goals</th>
</tr>
</thead>
<tbody>
<tr>
<td>“Plant overall safety and electrical systems are already included in the framework plan and the start of research in 2017 would be desirable. The Management Board has ordered small preliminary study projects on both topics in 2016.”</td>
<td>ESSI project responds to this new topic of the call based on the preliminary study carried out in 2016. ESSI will start the study focusing on electrical systems of NPP in order to improve the plant overall safety.</td>
</tr>
<tr>
<td>(Update to the Framework plan for 2017 call, p. 2)</td>
<td>(Update to the Framework plan for 2017 call, p. 2)</td>
</tr>
<tr>
<td>“There will be a change in the electricity production modes supporting the grid. The production that is dependent on weather conditions will increase and at the same time the production based on fossil fuel will decrease. Less conventional base load capacity will be available for maintaining the power balance and this may also have effects on the operation of the nuclear power plants. Potential safety issues related to the operation of nuclear power plants in the load following mode are an important topic for research.”</td>
<td>WP3 will work on the adaptation of NPP in the load following mode. The research will focus on the potential safety issues caused by adaptive operation of NPP to electrical systems and to national power grid stability.</td>
</tr>
<tr>
<td>(Update to the Framework plan for 2017 call, p. 1)</td>
<td>(Update to the Framework plan for 2017 call, p. 1)</td>
</tr>
<tr>
<td>“The changing technological environment presents new challenges for nuclear safety. New technical solutions require new methods for assessing a system’s behaviour and safety.”</td>
<td>A major energy system transition is underway in Europe. The strategy implies a substantial increase of the share of renewable, variable electricity production in the EU and also in Finland. WP3 assess the safety impacts on electrical systems and grid stability caused by adaptive operation of NPP.</td>
</tr>
<tr>
<td>(Ch. 3.2.2, p. 31)</td>
<td>(Ch. 3.2.2, p. 31)</td>
</tr>
</tbody>
</table>
"Assuring safety also involves preparing for very rare event chains, which naturally entail considerable uncertainties. The uncertainties can be managed as part of holistic safety planning and based on the risk and safety margins."

(Ch. 3.2.2, p. 32)

Open phase condition (OPC) is a source of common cause failure which may damage several or even all the redundant electrical power system parts of a NPP. During the last ten years, more than 10 cases of OPC in NPP have been reported by IAEA. WP1 analyses the OPC conditions and impacts and make recommendations about how to improve the security of NPPs in the case of OPC.

WP2 will examine what kind of overvoltages due to rare but extremely high lightning currents are able to cause in different parts of NPP electrical system. Possible recommendation for the coordination of overvoltage protection at different electrical system levels will be suggested.

A broad scale of various factors compromising safety or affecting the understanding of safety must be taken into account in safety assessment. Thus, at least the following must be included in the overall safety threats:

- faults within the plant: initiating events assumed based on various grounds, including both faults and functions that start without reason
- external threats to the plant, both natural phenomena and events related to human action

(Ch. 3.2.4.1, p. 33)

Lightning overvoltages can be caused in NPP power and automation systems via lightning strikes in the phase of the transmission line or lightning strike to the grounded parts of the NPP. The mechanisms how a very high lightning current is transmitted inside the NPP is not very well known. WP1 will give the probability of higher lightning currents and the impact they may have to the NPP electrical and automation systems.

"The topic is strongly connected to the systems engineering process and particularly to the requirements and information management and communication between the authorities, power company and supplier network. Part of the research may apply to the surveying and application in Finland of other fields and the practices in the nuclear field in other countries."

(Ch. 3.2.4.3, p. 35)

ESSI project is very closely connected to the information management and exchange between authorities, Finnish NPP companies and Finnish TSO Fingrid. In all WPs (WP1...WP3) will survey the applications and practises in Finland and in other countries with regard of open phase condition, lighting protection and adaptive operation of NPP.

".. the design of a nuclear power plant, one must prepare for various internal and external events that the plant must also cope with when faults occur. To prove this, various fault and effect analyses are carried out along with analyses on the behaviour of the plant as a consequence of the event under inspection"

(Ch. 3.2.4.4, p. 36)

WP1 analyses impact of open phase condition (OPC) on house load and electrical components (behaviour of protection, hazard of damage, urgency of mitigation actions in terms of time).

WP2 analysis of voltage stresses due to ground potential rise (GRP) in electrical, automation and control systems and possible mitigation means.
1.5 Education of experts

The project will increase understanding of the experts working with electric systems of NPPs in both effects of unbalance in OPC situations, and lightning overvoltage protection related issues. The ESSI project is used for education of younger expert in the field.
2. Work plan

The ESSI project is divided into three work packages each having a planned duration of two years (Figure 1). For each project year, a set of tasks will be defined with a clear scope and deliverables. While individual tasks may be short and their titles may be changed from the first year to the second one, they work on longer-lived topics within the work packages.

ESSI will have a multidisciplinary research strategy based on the research topics which were ranked the most important in the preliminary study for the electrical systems of NPP carried out in 2016. The purpose of WP1 is to build understanding of the different open phase condition cases and the consequences of possible voltage unbalances for different electrical system components, the criticality, risks and time required from mitigation actions. WP2 develops dedicated simulation methods and tools for assessing the safety in the case of lightning overvoltages and protection methods in different voltage levels of inside NPP electrical system. The idea of WP3 is to assess the operation of NPP in load following mode, its risks in future smart grid environment. Finally, project planning and progress reporting are allocated to a separate work package WP4 that also includes the necessary activities related to research collaboration, coordination and dissemination of results. As inkind work about three weeks are provided by power companies (one week from Fennovoima, Fortum and TVO) in order to give information of their preparedness to OPC in WP1 and data for overvoltage cases in WP2.

<table>
<thead>
<tr>
<th>WP 1 Open Phase Conditions</th>
<th>2017</th>
<th>2018</th>
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<tbody>
<tr>
<td>1.1 Analyses of unbalance conditions</td>
<td></td>
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<tr>
<td>1.1 Methods for detection of the OPC condition</td>
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<tr>
<td>1.2 Methods for monitoring of the components for OPC impact</td>
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<tr>
<td>1.3 Preparation of 2017 results for the development of protective measures and NPP operation guidelines</td>
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</tbody>
</table>

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<thead>
<tr>
<th>WP 2 Lightning overvoltage</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.2 Modelling of NPP electrical system for lightning strikes</td>
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<td></td>
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<tr>
<td>2.1 Modelling of ground potential rise in various systems of NPP and lightning strike locations</td>
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<td></td>
</tr>
<tr>
<td>2.2 Overvoltage protection of AC and DC system at lower voltage levels</td>
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</table>

<table>
<thead>
<tr>
<th>WP 3 Adaptive operation of NPP</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
<tbody>
<tr>
<td>3.1 Requirements for the maneuvering capabilities</td>
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<tr>
<td>3.1 Risks of adaptive control to NPP electrical systems and stability of the grid</td>
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</table>

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<thead>
<tr>
<th>WP 4 Project management</th>
<th>2017</th>
<th>2018</th>
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</table>

Figure 1. Tasks planned for 2017 in work packages 1 to 3 and their further planned continuation in the year 2018.
The planned 2017 tasks for the work packages WP1 through WP3 are listed in Figure 1, along with preliminary tasks considered for later research. The work packages and their tasks planned 2017 are described in detail in the sections below.

2.1 Open Phase Conditions (WP1)

In an open phase condition (OPC), one or two phases of transmission grid are disconnected and the voltages become unsymmetrical. Enabling cause of OPC is the fact that in transmission voltage levels, the circuit breakers (CB) are built as single phase units and thus it is possible that one or two phases remain open when all the three phases of the CB should be closed. OPC leads to unsymmetrical grid voltages, the severity of which depends on the transformer connections, how house load and generator are connected to the grid and on generator and network impedances. OPC impacts in a similar way all the network parts connected downstream of the phase disconnection point (i.e. point of common coupling, PCC). Thus it as a source of common cause failure which may damage several or even all the redundant electrical power system parts of a NPP. During the last ten years, more than 10 cases of OPC in NPP have been reported.

The impact of OPC depends on the severity of unbalance, which manifests itself in the form of negative sequence voltage and reduction in the positive sequence voltage. In generators, the negative sequence voltage causes vibrations, which may lead to mechanical damage. Generators are relatively sensitive to these and that is why they easily are tripped by unbalance relays. Tripping of generator, in turn, removes one strong symmetrical source of voltage from the network, and makes the unbalance even worse.

In directly connected motors, the negative sequence component of supply voltage causes torque which opposes the direction of rotation. At the same time, the positive direction voltage is reduced, and the machine takes correspondingly larger current in order to produce the mechanical power required. Sustained operation with unbalanced supply voltage may lead to either tripping of the motor protection, and hence disconnection of the motor from power supply, or in the worst case, damaging of the machine due to overheating.

If unbalance is sustained for a longer time, transformers and cables may be overheated due to excessive load currents. These components, however have relatively long thermal time constants and they are likely to be much more tolerant to OPC effects than motors.

In case of converters, the uninterruptible AC power system functionality can be impacted. Not only the bypass source may be degraded, but it may cause the rectifier/battery charger to shut down due to the input voltage being out of tolerance. The behaviour of these rectifiers depends on manufacture and settings of the device protections. However, if tripped, these devices usually return back to operation when voltages have been normalized.

The risks of OPC can be managed if there are means for timely warning of OPC situation, and if the NPP operation personnel is trained to take swift mitigating actions. The first condition can probably be satisfied by adding proper measurements and protective (alarming) relays in the NPP electrical network. The second condition requires knowledge about how severe is the situation and how much time the personnel has in their disposal to switch the house load to an alternative power source. Usually the NPP have several grid connection and power source options, which can be used as back up for each other. In the OPC situation it would be necessary to know, how these options are affected by the lost phase (phases) and which connections are available as they are in a sound condition.
It seems that OPC is a possible source of a severe common cause failure in a NPP, and hence it would be necessary to provide better understanding about the risks due to OPC and how these risks can be mitigated. The WP1 starts with following one task and objectives.

WP1 will be coordinated by Anna Kulmala (VTT). Partners and person months allocated to WP1 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>Aalto University</td>
<td>2</td>
</tr>
<tr>
<td>VTT Ltd</td>
<td>1,5</td>
</tr>
</tbody>
</table>

2.1.1 Analyses of unbalance conditions (T1.1)

OPC is a possible source of a severe common cause failure in a NPP, it is proposed that the WP1 starts by making a survey how the operating plants have been prepared to OPC and by a review of the existing literature. The task should take into account work already done e.g. by GRS (Gesellschaft für Anlagen- und Reaktorsicherheit) and continue from that including the following goals in 2017:

The survey how the operating plants have been prepared to OPC covers the protections against unbalance voltages in the case of disconnection of one or two phases of the transmission grid in the following two cases: Generator connected in the grid and generator disconnected. It is likely, that in case of severe unbalance in the grid, the generator is tripped by negative sequence voltage relay, which leads to the situation that NPP house load is supplied by the grid solely. This is probably the worst scenario from house load components point of view.

The literature review analyses also how possible locations of lost phases affect different alternative power supply sources (like 400 kV and 110 kV) and the unbalance in them (simultaneous OPC in both 400 kV and 110 kV). The impacts of OPC on various NPP loads and components are investigated. The goal is to assess the severity and time criticality of various unbalance cases and different components. It will be investigated impacts of OPC on the generator, with regard to an analysis of the operation of the unbalance protection and tripping of the generator in different OPC cases. Impact of OPC on house load and electrical components is focused on behaviour of protection, hazard of damage, and urgency of mitigation actions in terms of time. The loads studied are: direct connected motor drives, transformers, cables and rectifiers of AC/DC systems.

In 2018 the work will be continued with new tasks by investigating methods for detecting of the OPC conditions and for monitoring of the components for OPC impacts.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Anna Kulmala</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Antti Alahäivälä (Aalto), Riku Pasonen, Anna Kulmala (VTT)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>2 (Aalto ), 1,5 (VTT)</td>
</tr>
</tbody>
</table>
2.2 Lightning overvoltage (WP2)

Lightning can cause overvoltage in NPP inhouse power and automation systems via two different mechanisms: First, if lightning strikes in the phase of the 400 kV (or 110 kV) national transmission line connected to the NPP, an overvoltage can be transmitted through the transformers to the lower voltage levels. The house electrical system is protected against this kind of overvoltage by surge arresters, located on both sides of the main transformer, and usually also at the lower voltage levels like in medium voltage switchgears. The second mechanism is lightning strike to the shield wire or the tower of the transmission line. In this case, the lightning surge spreads into the grounding system, causing Ground Potential Rise (GPR) and, especially in the case of a very steep lightning current surge, potential differences between grounded parts and insulated power, communication and instrumentation circuits. A similar effect is caused by a lightning strike to the lightning rod (lightning conductor) used for the shielding of the NPP building, or if lightning ends up to the telecommunication mast or ventilation outlet mast of the power plant.

Lightning overvoltage impacts several equipment at the same time, and at least theoretically, it can be a triggering event of a common cause failure. The overvoltage protection is typically rated for 200 kA lightning strikes, which is considered to be the maximum expected lightning current. However, theoretically it is possible that this maximum current is exceeded, and it may be useful to examine the probability of higher lightning currents and the impact they may have to the NPP electrical and automation systems, either due to direct lighting strikes to the power system phase, or via GPR.

In the case of direct lightning strike, it should be examined which kind of overvoltages extremely high lightning currents (over 200 kA) are able to cause in different parts of NPP electrical system. The capacity of the surge arresters should be assessed and the probabilities of common cause faults on different voltage levels analyzed. One topic of interest, too, is the grading and coordination of overvoltage protection at different system levels. For instance, if at the high voltage side of a transformer there is flashover from phase to ground, the steep change of voltage is strongly connected through transformer capacitances to the secondary side.

With regard to the second overvoltage mechanism, the important factors are the construction and topology of the grounding system including how different electrical, automation and instrumentation systems are connected there, and the entrance point of lightning current in the grounding system. Of special interest also here are the lightning with extremely high currents and steep fronts. The task should be to study to GPR in different grounding system parts in order to investigate whether excessive overvoltages can be created between grounded and active parts of electrical, automation and control systems.

WP2 will be coordinated by Matti Lehtonen (Aalto). Partners and person months allocated to WP2 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>Aalto University</td>
<td>5</td>
</tr>
</tbody>
</table>

2.2.1 Modelling of NPP electrical system for lightning strikes (T2.1)

In 2017, the lightning overvoltages coming from the national grid to the NPP electric systems are investigated. The research is focused on extremely high current surges. The data of the magnitude and probability for the surges are available in Fennovoima’s internal report collected from FMI’s statistics (FMI= Finnish Meteorological
In this task T2.1, the modelling of NPP electrical system is done for lightning strikes coming from the 400 kV or 110 kV national grid. The cases analysed are overvoltage due to direct lightning to the phase and back-flashovers close to the grid transformer. The latter case produces very steep overvoltages which may have a strong capacitive coupling through the transformer.

In addition to the modeling of overvoltages, the work comprises analysis of the present capacity of surge arresters and analysis of overvoltages created in lower system levels if flashover at the high voltage side of grid transformer. The result is possible recommendations for the coordination of overvoltage protection at different electrical system levels.

In 2018 the work will be continued by the analysis of GPR in cases where lightning strikes in various grounding system parts, and analysis of voltage stresses due to GPR in electrical, automation and control systems and possible mitigation means of mitigation of the GPR overvoltages. Also new task will start regarding overvoltage protection of AC and DC systems at lower voltage levels in the plant’s internal grid in 2018.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Matti Lehtonen (Aalto)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Mohammad Rizk (Aalto)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>5 (Aalto )</td>
</tr>
</tbody>
</table>

### 2.3 Adaptive operation of NPP (WP3)

Conventional wisdom holds that nuclear generation is inflexible, because nuclear power plants are mostly operated at constant full power. This has been mainly due low variable (i.e. power level dependent) costs of nuclear, making it the cheapest baseload generation option. Technically, nuclear power plants are capable of large and rapid output variation: in Germany and in France the operating nuclear plants perform significant power adjustments on daily basis. Large-scale utilization of this capability requires supporting research to determine the costs of increased wear and tear as well as the benefits of more flexible operation that can be accrued from the power and ancillary service markets.

There is an increased need for flexibility in the future power system due to fluctuating power generation especially from wind and solar power plants. There have been some preliminary studies on how to use Finnish nuclear power plants in partial power. As the share of variable power generation is set to increase strongly in Northern Europe, there is a need for a more extensive study detailing different benefits, problems and risks for part-loading Finnish nuclear power plants in the future. Nuclear power plants experience similar issues from part-loading as other thermal power plants. Changing heat gradients cause wear and tear in the components, which can increase the need and costs of maintenance as well as a possible reduction in the lifetime of the equipment. In addition, nuclear power plants have distinct characteristics due to nuclear safety issues and neutron absorption dynamics that need to be accounted for. The experience from variable operation of nuclear power plants in France and Germany will be used as a reference to understand the possibilities for part-loading Finnish nuclear power plants.

The need and potential benefits of part-loading nuclear generation will depend on the characteristics of the future power system. There is naturally high level of uncertainty how much variable power generation will actually be built and what kind of flexibility resources as well as markets and regulations there will be to cope with the increased variability and uncertainty.
The objective of the WP3 is to estimate the technological limits and risks of adaptive control in today’s nuclear power plants with regard to electrical systems in order to avoid the increase of disturbances in power plant.

WP3 will be coordinated by Riku Pasonen (VTT). Partners and person months allocated to WP3 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months in 2017</th>
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<tr>
<td>VTT Ltd</td>
<td>2</td>
</tr>
</tbody>
</table>

2.3.1 Requirements for the maneuvering capabilities (T3.1)

Nuclear power plants in Finland are normally not operating in manoeuvring regimes. However, the more we get stochastic generation and in the future also stochastic or movable consumption and energy storages in the form of cars, the more important this issue becomes. In fact, most of the currently operating NPPs were designed to have strong manoeuvring capabilities (NEA, 2011). However, operating an NPP at a constant power level is simpler and less demanding on the plant’s equipment and fuel. Usually three types of manoeuvring are defined: primary and secondary frequency regulation (which depend on current grid demand) and predefined variable load programmes (i.e. reductions or increases in power output agreed in advance with the grid operator). Planned reductions or increases in power output allow initial balancing of electricity supply and demand. These variations can be significant.

Adaptive operation of NPP may have risks for electric systems and control inside NPP and for stabilization of national grid. It is expected that the disturbance sensitivity of the electrical components and ICT systems can increase. Load following operation has some influence on the ageing of certain operational components and thus one can expect an increase in maintenance.

The objective is to estimate the technological limits and risks of adaptive control in Finnish nuclear power plants with regard to electrical systems in order to avoid the increase of disturbances in power plant. In the year 2017, the minimum requirements for the manoeuvrability capabilities of Finnish nuclear power plants will be mapped and clarified. The mapping will base on the utility requirements from the safety point of view. This includes also the manoeuvring capabilities of Finnish NPPs in different types of frequency and balancing control of the national grid. During this task information will be exchanged with the GINO – Grid Interference on Nuclear power plant Operation- project, Energiforsk in Sweden.

The work will be continued in 2018 by estimating technological limits and risks of adaptive control in Finnish nuclear power plants with regard to electrical systems.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Riku Pasonen (VTT)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Anna Kulmala, Riku Pasonen, Seppo Hänninen (VTT),</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>2 (VTT)</td>
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</table>
2.4  Project management (WP4)

2.4.1  Management and reporting (T4.1)

VTT will act as the project coordinator, managing communication with the project’s research group, other projects, the reference group guiding the project and the programme management, and is responsible for the reporting obligations set for the projects in the programme. Senior Scientist Seppo Hänninen will act as the VTT project manager.

A co-operation agreement between ESSI project group members will be signed. The project managers of each member organisation (see Chapter 4.2) will share the responsibility for the actualisation of the research objectives.

T4.1 will also include the preparation of the project plan for 2018.

Project coordination will be carried out according to VTT practices. VTT’s operational system has been certified in accordance with the ISO 9001:2008 standard. The certificate was issued by DNV in 2006, and covers research, technology transfer and consultation services and the development of new technology at VTT.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Seppo Hänninen (VTT)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Seppo Hänninen (VTT)</td>
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<tr>
<td>Person months in 2017</td>
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## 3. Deliverables and milestones 2017

<table>
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<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
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<tr>
<td>D1.1.</td>
<td>Unbalances caused by different OPC cases in NPP electric systems and preparedness of operating plants against OPC. Project report</td>
<td>3,5</td>
<td>9/2017</td>
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<td>D2.1</td>
<td>Modelling and simulation of the lightning overvoltages penetrating from the 400 kV or 110 kV grid to the NPP electric systems. Scientific Conference paper</td>
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<td>9/2017</td>
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<td>D3.1</td>
<td>Requirements for the maneuverability capabilities of Finnish nuclear power plants. Project report</td>
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4. Project organisation

4.1 Project management

The organisation responsible for the coordination of the whole project will be VTT. Senior scientist, Dr Seppo Hänninen will act as the project manager at VTT. He has 3 year experience in electromechanics at TKK, 7 years experience with Electrical Inspectorate on electrical inspection activities and 25 year experience with VTT on electricity distribution, smart grids and reliability engineering.

In Aalto, the head of the project is professor Matti Lehtonen, who has 18 year experience of electric power systems as the professor in charge of teaching and research in the field “Power Systems and High Voltage Engineering” at Aalto University (formerly TKK). Prior to the University, he was 12 years with VTT, working on electrical distribution systems.

4.2 Project consortium

The participating organisations (and the managers of the organisation-specific work) are:

1. Aalto University (Professor Matti Lehtonen)
2. VTT Technical Research Centre of Finland Ltd (Senior Scientist Seppo Hänninen)

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
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<tr>
<td>Antti Alahäivälä</td>
<td>Research scientist</td>
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<td>Mohammed Rizk</td>
<td>Research scientist (PhD)</td>
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<td>Seppo Hänninen</td>
<td>Senior Scientist (PhD)</td>
<td>VTT Ltd</td>
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<td>Riku Pasonen</td>
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<td>Anna Kulmala</td>
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5. Risk management

The identification, evaluating and managing risk base on the methodology that VTT uses in project preparation. Significant and intolerable risks are printed in red in table below. Contingency plan for these risks are presented in 5.2.

5.1 Identification and evaluation

<table>
<thead>
<tr>
<th>Likelihood</th>
<th>PROJECT RISK EVALUATION</th>
<th>Impact / consequences</th>
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<td></td>
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<td>SERIOUS (3)</td>
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<td>PROBABLE (3)</td>
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<tr>
<td></td>
<td>4 Significant risk</td>
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</tr>
<tr>
<td></td>
<td>5 Intolerable risk</td>
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<tr>
<td>POSSIBLE (2)</td>
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<tr>
<td></td>
<td>3 Moderate risk</td>
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</tr>
<tr>
<td></td>
<td>4 Significant risk</td>
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</tr>
<tr>
<td>UNLIKELY (1)</td>
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<tr>
<td></td>
<td>2 Low risk</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3 Moderate risk</td>
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<p>| Project objectives, definitions, tasks, maturity of the relevant technology |
|-------------------------------|-------------------|------------------|
| T 2.1 Modelling of NPP electrical system for lightning strikes | 3 | 2 | 4 |</p>
<table>
<thead>
<tr>
<th>Issue</th>
<th>1</th>
<th>2</th>
<th>3</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>External mandatory rules and regulations (customer, legislation, customs rules)</strong></td>
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<tr>
<td><strong>Project’s human resources</strong></td>
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<td>3</td>
</tr>
<tr>
<td>Management of time usage.</td>
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<td>2</td>
<td>3</td>
</tr>
<tr>
<td><strong>Timetable or cost pressures and financing</strong></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>The project has challenging timetable and cost targets.</td>
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<td>3</td>
<td>4</td>
</tr>
<tr>
<td>Possibilities/ limitations to align timing (etc.) with other unfinished or planned projects.</td>
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<td>2</td>
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<tr>
<td><strong>Project’s external stakeholders / cooperation with subcontractors</strong></td>
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<tr>
<td>Match of the organization cultures.</td>
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<td>Challenges to get detailed input data for simulation modelling and analysing the inside electrical systems</td>
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<td>3</td>
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<tr>
<td><strong>Equipment, premises and infrastructure (project/ research environment)</strong></td>
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<td>Possible equipment failures that may lead to interruptions or delays in the research work</td>
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<tr>
<td><strong>Occupational safety, environmental and information security risks</strong></td>
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<td>Information security risks (computers/ PCs, tablets, smartphones, tele-communication, inadequate security culture etc.)</td>
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<td>General employee safety risks (traveling, hazards related to geopolitical circumstances/ political unrest etc.)</td>
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### 5.2 Risk management actions for significant and intolerable risks (risk class 4-5)

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<thead>
<tr>
<th>Risk name</th>
<th>T 2.1 Modelling of NPP electrical system for lightning strikes</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Description and impact:</strong></td>
<td>The estimation of the ground potential rise is very challenging task and may have impacts of the results of this task in order to estimate the risks for power system components and the mitigation methods.</td>
</tr>
<tr>
<td><strong>Evaluation date:</strong></td>
<td>Contingency plan and decisions</td>
</tr>
<tr>
<td><strong>Person responsible:</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Schedule:</strong></td>
<td></td>
</tr>
<tr>
<td>Regular follow-up of the progress of WPs and resource use.</td>
<td>Project manager</td>
</tr>
<tr>
<td>End of year 2017</td>
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</table>

<table>
<thead>
<tr>
<th>Risk name</th>
<th>The project has challenging timetable and cost targets.</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Description and impact:</strong></td>
<td>The objectives of the project are very ambitious and they cannot be reached within the proposed timeframe and with the allocated resources leading to delay of deliverables.</td>
</tr>
<tr>
<td><strong>Evaluation date:</strong></td>
<td>Contingency plan and decisions</td>
</tr>
<tr>
<td><strong>Person responsible:</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Schedule:</strong></td>
<td></td>
</tr>
<tr>
<td>Early discussion with contact persons on appointment issues.</td>
<td>Project manager</td>
</tr>
<tr>
<td>First check point in summer 2017</td>
<td></td>
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</tbody>
</table>
References


Ikäheimo Jussi & Kiviluoma Juha. Operating P2G in a power system with large amounts of PV, wind power and hydro power. 9th International Renewable Energy Storage Conference (IRES 2015), 9-11 March 2015, Dusseldorf, Germany.

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SAFETY REPORT (DD 1142) Impact of open phase conditions on electrical power systems of nuclear power. IAEA SAFETY REPORT-V.05 10 April 2015. Received from STUK. 57 p.

Smed 2015 Thomas Smed, Robust electric power in the Forsmark power plant. Powerpoint presentation. IAEA open phase day, May 2016.


## Electric Systems and Safety in Finnish NPP

### Expenses

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<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat&amp;supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Memb fee</th>
<th>Other</th>
<th>TOTAL</th>
<th>VYR</th>
<th>Fennovoima</th>
<th>Fortum</th>
<th>TVO</th>
<th>Aalto</th>
<th>LUT</th>
<th>VTT</th>
<th>NKS</th>
<th>Other</th>
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<tr>
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### Financing

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<th>Volume</th>
<th>Personnel</th>
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### Comments:

- T2.1, T3.1: Participation in the ICNPSE 2017 or similar conference and information exchange with GINO – Grid Interference on Nuclear power plant Operation project.
- T1.1 and T2.1 and T3.1: In-kind work about three weeks (one week from each power company, Fennovoima, Fortum and TVO) in order giving input data for modelling.
- T1.1, T2.1: Domestic travelling to Olkiluoto and Lovisa.

Memb fee: Please explain the membership fees in international projects included in the cost budget (VYR funding 100% if approved).

Describe possible in-kind work contribution here (organisation and person months, use of equipment etc.).

Other explanatory comments.
### Expenses & Financing Table

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**Comments:**

- **T2.1:** Participation in the ICNPSE 2017 or similar conference and information exchange with GINO – Grid Interference on Nuclear power plant Operation-project, 1.5 k€
- **T2.1:** Inkind work about three weeks (one week from each power company) in order give input data for modelling
- **T2.1:** Domestic travelling to Olkiluoto and Loviisa, 0.5 k€

Memb fee: Please explain the membership fees in international projects included in the cost budget (VYR funding 100% if approved). Describe possible in-kind work contribution here (organisation and person months, use of equipment etc.).
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- T1.1, T3.1 Inkind work about three weeks (one week from each power company, Fennovoima, Fortum and TVO) in order giving input data
- T1.1 domestic travelling to Olkiluoto and Loviisa.

Memb fee: Please explain the membership fees in international projects included in the cost budget (VYR funding 100% if approved).

Describe possible in-kind work contribution here (organisation and person months, use of equipment etc.).

Other explanatory comments.
SAFIR2018 Project plan

**EXWE**

Extreme weather and nuclear power plants

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Havu Pellikka, Ulpu Leijala, Jan-Victor Björkqvist, Jani Särkkä, Milla Johansson

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1. Research theme and motivation

Overall safety management over the life cycle of a nuclear power plant (NPP) requires, among others, evaluation of external events triggered by exceptional weather and sea-level conditions (YVL B.7, 2013). Measures against impacts of adverse weather events are already taken into account in the design phase of new power-plant units, such as the one under construction at Olkiluoto (OL3) and the other in planning at Hanhikivi. In the existing Finnish NPP units, preparedness against extreme natural phenomena is continuously being improved. For example, technical solutions and systems have been modified at OL1 and OL2 in order to reduce the risk of heavy rainfall-induced flash floods in the yard area. Moreover, actions have been taken to prevent a blockage of air intake of emergency diesel generators as a consequence of simultaneous snowfall and wind. Furthermore, a means of preventing problems due to frazil ice formation are applied both at Olkiluoto and Loviisa NPPs. (Viitanen et al. 2013). As a fourth example, backup systems of the Loviisa NPP have been supplemented by air cooling towers that are independent of the seawater normally used to cool the plant’s reactors. (Fortum 2015)

Despite the already-taken measures against impacts of exceptional natural phenomena on the NPPs, it is essential to deepen the scientific knowledge on exceptional external conditions and on the occurrence of cascading (multiple) events in particular. This is because estimates of frequencies of weather-related and sea-level-related hazards, used as design basis and in the Probabilistic Risk Assessment (PRA) of NPPs (YVL A.7, 2013), are subject to considerable uncertainties. The main challenge in evaluation of frequencies of extreme phenomena arises from their nature: these events occur very rarely or even are unprecedented in Finland and they often only have a small spatial and temporal scale. Consequently, the measured time series, if available at all, tend to be quite restricted. Equally importantly, the probability of occurrence of exceptional external conditions around the NPP sites is subject to global climate change. Therefore, the patterns of extreme weather events—both in frequency, extremity, and magnitude—are likely to alter in the course of time. A hazard curve evaluated from time series of past measurements needs to be regularly updated. Beneficially, the updates are empowered by recent accretions of research material and developments of methods.

The EXWE project primarily focuses on localized 1) extreme weather and 2) extreme sea-level events that affect the design principles of the power plants and might pose external threats to the plants. In addition, the project aims to 3) provide a modern platform for atmospheric dispersion modelling of accidental releases. Like the first two topics, also the third theme deals with the environment of a nuclear facility, but now impacts of the plant on the environment are considered rather than vice versa. An additional topic, potential space weather effects on nuclear plants, was addressed in 2015-2016 but based on comments by reviewers of the original research plan for 2017, the topic was excluded from the current revised research plan.

1.1 Background and state-of-the-art

Extreme weather events

Extreme convective weather (hereafter called ECW) – whether caused by mesoscale convective systems (such as derechos, in Finnish “syöksyvirtausparvi”), detached thunderclouds or convective elements in extra-tropical cyclones – may materialize as heavy rain, large hail, intense lightning, strong wind gusts (i.e., downbursts) or tornadoes (termed as waterspouts over sea, and in Finnish “trombi” or “vesipatsas”). ECW may result in flash floods and coastal flooding, including also meteotsunamis. While summertime ECW is more likely to occur, wintertime sea-effect ECW occasionally develops over ice-free sea areas and, depending on the mean wind direction, may lead to excess coastal snowfall.

Development of ECW requires specific atmospheric circumstances. Increasing observation density owing to the use of weather radars, satellites and lightning-location sensors has revealed that convective storms are more common in northern Europe than generally expected (see Romero et al. 2007, Tuovinen et al. 2009, Saltikoff et al. 2010a, Rauhala et al. 2012, Mäkelä et al. 2014, Funkka et al. 2015 for summer ECW and Saltikoff et al. 2010b, Niemelä 2012, Mazon et al. 2015 for winter ECW). However, research on linkages between the extreme small-scale phenomena and larger-scale (synoptic and mesoscale) conditions has mostly conducted in warm or mild
rather than in cool-climatic regions of the world. Therefore, more efforts are needed in order to combine all available information about the past occurrence of ECW in Finland. At the same time, one needs to consider the issue of how to do the coupling and transfer between the different observational and modeled datasets, as discussed in EXWE in 2015 (Hyvärinen et al. 2015).

Partly due to the gaps in the understanding of the present ECW climate in Finland, the question of how severe convective weather will change in the future is an open one. The connection between convective extreme weather and climate change has recently been studied by Brooks (2012), concentrating on the USA, Marsh et al. (2009) and Pucik et al. (2016), addressing Europe, Vajda and Rauhala (2011) focusing on Finland, and Haarsma et al. (2013) addressing the Euro-Atlantic sector. Based on Brooks (2012), atmospheric conditions favorable for severe thunderstorms could become more frequent in the future, possibly also in Europe. Progress beyond the state-of-the-art would be achieved in this project by tailored downscaling of up-to-date climate model data.

Typical conditions of wintertime extreme convective weather or sea-effect snowfall include 1) ice-free sea surface which acts as a constant source of energy and moisture into the atmosphere, 2) cold air mass over the sea and 3) wind which blows from a suitable direction over the sea. When these prerequisites are met, the vertical temperature difference and the moisture flux from the sea surface generate strong rising air motion and convective precipitation. Over the Gulf of Finland, a sea-effect snowfall case can typically take place when the sea is ice-free and very cold continental air mass flows to the region from the east (Niemelä 2012, Savijärvi 2012). In the future, wintertime ice-free conditions of the Baltic Sea are likely to become more common (Luomaranta et al. 2014), but the fraction of solid precipitation is projected to decrease (Räisänen and Eklund 2012; Räisänen 2016). Therefore, it is not evident whether extreme sea-effect snowfall cases will become more severe or frequent or vice versa.

Besides intense snowfall, freezing rain (and rime) may increase the risk of blockage of air intakes, depending on the NPP site (Tietäväinen et al. 2012). Freezing rain is a topic about which there is little scientific literature available in Europe. An optimal vertical temperature profile for freezing rain consists of subzero temperatures near the ground and a melting layer aloft (Carrière et al., 2000; Laine, 2008). Consequently, the sign of future trends in occurrences of freezing rain depends on whether such vertical temperature structures will occur more or less often during future precipitation events (see Cheng et al. 2007, Lambert and Hansen 2011 for North America). In order to assess the occurrence of freezing rain in the future on the basis of climate model data, the first step in EXWE was to develop a method to detect freezing rain from gridded reanalysis data, i.e. from reconstructions of past atmospheric conditions (Kämäräinen et al. 2017). The work was partially carried out in international projects RAIN (Risk analysis of infrastructure Networks in response to extreme weather) and CLIM4ENERGY (A service providing climate change indicators tailored for the energy sector).

The proposed project would bring out advance in frequency evaluations of extreme warm- and cold-season convective weather and freezing rain at the NPP sites by utilizing recent developments of numerical atmospheric and climate modelling, together with archives of remote sensing and meteorological reanalysis data and more traditional types of observations.

**Extreme sea level events**

Extreme sea levels are crucial for the safety of NPPs often located at coastal areas. Previous research in EXWE has produced i) scenarios of mean sea level changes on the Finnish coast, based on global sea-level rise scenarios and their influence on the Baltic Sea, ii) probabilities of extreme sea levels based on long-term tide gauge observations carried out for more than 100 years on the Finnish coast, and iii) most recently, knowledge on meteotsunamis, tsunami waves created by atmospheric processes.

In previous SAFIR programmes, the focus within EXWE has been on long-term, conventional (hourly) sea level data. During SAFIR2018, the research extends towards new high-resolution data giving a more detailed picture on short-term sea level variations. Research topics include complementing the distributions of extreme sea levels with the effect of wind waves, and analysing extreme events, including meteotsunamis, in recent high-resolution (1-min) sea level data. The aim is to combine the existing knowledge of long-term variability with short-timescale processes that have a potential to further increase the damage of extreme-sea-level events. Combining long- and short-term sea level phenomena is essential in the second phase of sea-level risk estimates.

Global research around the topics of sea level rise and meteotsunamis is active but cannot be applied to Finland as such because of the special characteristics of the Baltic Sea and large regional anomalies in sea-level rise (Johansson et al. 2014). For example, meteotsunamis are a recently recognized hazard in Finland following recent observations on the Finnish coast (Pellikka et al. 2014). Their formation process is relatively well understood (Monserrat et al. 2006) and meteotsunamis are subject to active research in different parts of the world, especially in the Mediterranean (Vilibić et al. 2014). The research in EXWE aims at results which are directly applicable to the Finnish NPPs: we utilize the knowledge gained in other areas of the world while taking into account local conditions.
Main results of EXWE in 2015 for extreme-sea-level events included: i) extending the time series of meteotsunami occurrence in the Gulf of Finland by one decade (1980–1989), ii) first estimates of hazard curves (probability distributions) of meteotsunami occurrence at the NPP sites, and iii) developing a statistical method to evaluate the joint effect of high sea level and high waves, by combining the probability distributions of the two phenomena. This method development will result in improved flood probability estimates in the future.

In 2016, the research concentrated on three different topics: i) analyzing 10 years of high-frequency (1-minute) sea level data from the Finnish coast, identifying extreme events from the data and analyzing their meteorological background, ii) further testing and developing the method to estimate the joint effect of sea level and waves, and iii) a literature review on sea level rise and ice sheet instability.

In 2017, the efforts to further develop the improved method of estimating flooding risks will continue, together with a wave model validation study to improve our ability to estimate the distribution of wind waves (for which there is often not enough observational data). A new research topic assessing the highest sea levels that could theoretically occur on the Finnish coast will be started in 2017, to be continued in 2018. Finally, the meteotsunami data obtained in EXWE in previous years will be summarized and offered for review to the scientific community.

### Integrated dispersion and dose assessment

Transport and dispersion of pollutants from a source located at the coast are subject to flow patterns and structures related to the contrasting aerodynamic roughness and thermal inertia of land and sea surfaces. Important coastal wind systems include the sea-breeze circulation blowing on shore at the surface and off shore aloft, and low-level jets blowing along the coastline. The atmospheric boundary layers over land and over sea are often quite different in height, stability, and turbulence; and the flow across the coastline is then accompanied by a more-or-less marked thermal internal boundary layer (TIBL).

The dispersion calculations performed for nuclear safety assessments are typically based on Gaussian dispersion modelling. However, this assumption limits such studies to the range of 10–20 km from the site, and moreover, the Gaussian models are not well suited for simulating dispersion driven by highly dynamical and spatially complex mesoscale weather systems. Meanwhile, Lagrangian and Eulerian dispersion models, such as the FMI’s SILAM model (Sofiev et al., 2006, 2015), have been developed for operational use in combination with numerical weather prediction models. In parallel, meteorological high-resolution modelling has progressed to a stage where a proper assessment of coastal meteorological conditions can be fed into the dispersion modelling system. Introduction of high-resolution (up to 0.5 km) weather prediction systems opens up opportunities for high resolution dispersion modelling which resolves explicitly both mesoscale (convection, land-sea breezes) and large scale meteorological features.

In a parallel work to EXWE, FMI has Integrated state-of-art exposure modules to SILAM in 2016. The implemented new dose-assessment modules for SILAM dispersion modelling system are based on the experience with the modern-type risk assessment that has been gained by FMI within the recent risk assessment project for a potential NPP in Lithuania. The tool is explicitly connected to the SILAM dispersion model and represents, from a technical standpoint, a post-processor to the SILAM output. The approach and functionality of the tool, albeit limited to personal dose assessment, has passed the scrutiny of an expert evaluation of an international expert panel that was assessing the report of the project.

These state-of-art integrated modelling tools for dose assessment and meteorological modelling developed within FMI research and application projects, as well as during the routine preparedness to the emergency response, are considered as a starting point for the work within this project. The aim is to develop an integrated dispersion and dose assessment toolset based on connecting the SILAM dispersion model with state-of-the-art dose-assessment software. The main results in 2015 were related to the case study planning, as well as to designing the new dose as-sessment tool, its connection with the SILAM dispersion model, and relation with the VALMA tool currently used by VTT. In 2016, updates of the HARMONIE high-resolution meteorological model have been carried out in support of calculations by the SILAM dispersion model, dedicated new simulations by HARMONIE are being analyzed, and a first version of a new SILAM graphical user interface (GUI) is being developed.

### 1.2 Objectives and expected results

The objective of the research is to enhance scientific understanding of the environmental conditions of the NPP locations and predicting how they can change. By clearly explaining the methods and dataset fusions we enable replicability of the work and increase reliability in the results. EXWE primarily focuses on extreme weather and sea-level events that affect the design principles of the power plants and might pose external threats to the plants. A specific focus is given to extreme warm- and cold-season convective weather, including tornadoes and downbursts; wind-related multiple events, freezing rain; and extreme-sea-level events such as meteotsunamis. The work focus-
ing on atmospheric dispersion modelling aims to provide a modern platform for assessing consequences of accidental releases at multiple transport and time scales. A spectrum of different sources of information will be utilized as the research material.

The expected results include:

- Frequency and trends of extreme convective weather conditions near the coasts of Finland.
- Frequency and trends of freezing rain at the NPP sites near the Finnish coast.
- Frequency and trends of combined weather events related strong winds near the Finnish coast.
- Exceedance frequency distribution of meteotsunamis at the Finnish NPPs.
- Combined probability distribution of high sea level and high waves on the Finnish coast.
- Follow-up of new knowledge regarding sea-level rise and, when necessary, updates to the previously calculated sea-level scenarios on the Finnish coast up to 2100.
- An integrated dispersion and dose-assessment toolset (SILAM Dose Assessment Post-Processor DAPP).

### 1.3 Exploitation of the results

The results of the project can be used to improve the design of future nuclear plant units and to further improve the safety of existing units against the effects of natural phenomena. The end-users are 1) the power companies designing and running power plants, and 2) the nuclear safety authorities defining the safety regulations for NPP constructions and operations.

Concrete examples of purposes to which the results of the proposed project could be exploited are given in a paper by Tietäväinen et al. (2012), acknowledging experts in the adhoc group of the previous EXWE/SAFIR2011 project. The paper shows a compilation of the external risks related to sea-level rise, extreme weather events and other local geophysical phenomena from the viewpoint of the design and operation of a nuclear power plant. Regarding extreme convective and freezing weather phenomena, issues of concern for NPP safety include loss of offsite power, isolation of the plant and blocking of air or water intakes, depending on the NPP unit. Lightning might also cause grid disturbances; high peak current failures and over-voltages. Hurricane-force or almost hurricane-force winds, like those blowing in Scotland in 2011 and 2014 (MetOffice, 2011, 2014), might even cause damage to buildings. If the crest height of a meteotsunami wave would rise above the design basis, flooding of safety-critical compartments could occur, and if the seawater had time to penetrate into the buildings, there could be severe consequences.

In addition to publishing the new research results in peer-reviewed scientific journals, we plan to disseminate the results in a well understandable format, aiming at a high relevance of the project results for nuclear safety. Science news will be released by FMI when the manuscripts are accepted and in print. Climateguide will be used to disseminate the information in Finland. We plan to start to organize annual or biennial seminars of extreme climate research at FMI. We also present our progress in international conferences.

### 1.4 Appropriateness of the project to SAFIR2018 programme

The project is appropriate to SAFIR2018 programme because its topics are related to overall safety management, the concept of defence-in-depth (DID), Probabilistic Risk Assessment (PRA) and modelling of environmental impact. The SAFIR2018 framework plan, under the research area “Plant safety and systems engineering”, emphasizes the importance of better understanding and new information of natural hazards, such as storms and other extreme weather events (Sec. 3.2.4.4 for factors influencing technical solutions, page 38). Under the research area “Reactor safety”, it is stated in the framework plan that PRA level 2 computation still requires the transfer of competence, development and research (Sec 3.3.4.7 for Probabilistic risk analysis). It is also stated there that in connection with the licensing of plans, it must be possible to make independent estimates of the spread of radioactive releases into the atmosphere (Sec. 3.3.4.8 for Modelling of environmental impact). The EXWE project, dealing with two-way influences between the environment and power plants, aims to improve the evaluation of extreme geophysical events posing threats to power plants and to improve the modelling of the spread of radioactive releases into the atmosphere. Besides, the project educates new experts and aims to produce peer-reviewed journal articles, as shown later.

EXWE also contributes to international networking through extensive worldwide connections to other meteorological and oceanographic institutes and expert groups, such as the European Severe Storms Laboratory (ESSL), the National Severe Storms Laboratory (NSSL) in the US, and the Institute of Oceanography and Fisheries in Split, Croatia. In 2015-2016 EXWE co-operated with the EU FP7 project RAIN that contributed to minimizing the impact of extreme weather events on transport, energy and telecommunication networks and, among others, considered the Loviisa NPP during the flooding of January 2005. In 2016-2017, work related to freezing rain is being partially carried out within CLIM4ENERGY, a project contributing to the Copernicus Climate Change Service (C3S). The
non-hydrostatic convection-permitting HARMONIE model, utilized both in WP1 and WP3, 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. Finally, EXWE’s work in 2015-2016 has been partially funded by the Swedish Radiation Safety Authority (SSM). Thereby the proposal supports the goals of the programme, as well as its impact indicators given in the SAFIR2018 Operational management handbook.

1.5 Education of experts

The project trains new experts to the nuclear power plant safety area, in the wide sense of the defence-in-depth principle, by enhancing know-how of geophysicists on hydro-meteorological hazards of relevance to the overall safety of NPPs. The following theses and dissertations are expected:

- Jenni Rauhala: PhD dissertation in 2017 (tornados, thunderstorms)
- Matti Kämäräinen: PhD studies (climate models, proxies and downscaling)
- Anna Luomaranta: PhD dissertation in 2018 (snow- and sea ice-related multiple events)
- Taru Olsson: PhD dissertation in 2019 (climate models, bias correction, extreme precipitation)
- Tuuli Perttula: PhD in 2018 (data assimilation of remote sensing data into meteorological models)
- Havu Pellikka: PhD dissertation in 2017 (meteotsunamis, sea level scenarios)
- Ulpu Leijala: PhD dissertation in 2018 (sea level research, flooding risks)
- Jan-Victor Björkqvist: PhD dissertation in 2017 (wave research)
- Julius Vira, PhD dissertation in 2017 (data assimilation in atmospheric dispersion modelling)

Ulpu Leijala makes a 2-month research visit to NERSC (Nansen Environmental and Remote Sensing Center) in Bergen, Norway in early 2017, to collaborate in sea level and flooding risk research.
2. Work plan

2.1 Extreme weather (WP1)

The work of extreme weather aims to improve the reliability and accuracy in assessments of extreme weather events, both in frequency and magnitude. Research focusing on past cases of extreme weather is the backbone for the estimations of severe-weather occurrence in future climates. The difficulty lies in the fact that the studied weather phenomena typically have temporal and spatial scales smaller than what can be covered by the conventional weather station network and resolved by climate models. Therefore additional information about them and their impacts need to be gathered and advanced methods are required in order to be able to produce a comprehensive view about their probability of occurrence in the past and to assess influences of climate change in the future. Our procedure is outlined in Fig. 1 and described below in some more detail.

1) Information about past extreme-weather cases has been collected from various sources of observations (for previous EXWE deliverables, see Mäkelä et al., 2015; Laurila et al. 2016).

2) Using the actual past cases, factors, thresholds or patterns are being identified that favour or can trigger extreme weather. Meteorological reanalysis data for the identified past cases are utilized, together with simulations with a very-high-resolution non-hydrostatic atmospheric model. (For previous EXWE deliverables focusing on sea-effect snowfall and thunderstorms, see Luomaranta et al., 2015; Olsson et al. 2016; Ukkonen et al., 2016).

3) Using reanalysis data throughout the past decades and the previously defined factors or thresholds as predictors or proxies, we can get hints of potential past extreme cases in Finland. The candidate cases are further examined for verification of the indicators of favourable atmospheric conditions, after which the frequency and trends of the phenomena in the past can be assessed. (For previous EXWE deliverables, see For freezing rain, see Luomaranta et al. 2015; Kämäräinen et al., 2017)

4) Using the verified indicators of atmospheric conditions that support development of extreme weather, but now together with climate model data for the future instead of reanalysis data for the past, we aim to estimate how the frequency and intensity of the extreme weather events may change during the 21st century. (For freezing rain, see Kämäräinen et al., 2016)

![Figure 1: A schematic of the procedure used in EXWE to study the occurrence of severe weather events and to assess climate change impacts on them.](image)

In 2017, our objective is to increase the solidity of estimates of probabilities of extreme warm-season convective weather phenomena, intense sea-effect snowfall and severe freezing rain conditions in Finland, and at the NPP sites in particular, by deepening our understanding of the occurrence of these events and modelling aspects of them.

Warm-season convective weather phenomena. Following a review in 2015 about the present status of knowledge regarding warm-season extreme convective weather (ECW), historical time series of various ECW phenomena were completed and combined in 2016. Notable cross-correlations were found between annual numbers of i) thunderstorm days, lightning density and heavy rainfall, ii) heavy rainfall and tornadoes, and iii) large hail and tornadoes. In addition, the average annual number of thunderstorm days appeared to have clear decadal variations. While the data set of lightning in Finland is robust and long, dating back in time to 1887, the time series of the other ECW phenomena, especially those needing human observations (tornadoes and large hail), are likely to miss observations. Accordingly, an important question is how well the occurrence of hail or lightning can be used as a proxy for the other hazardous ECW events related thunderstorms (and for atmospheric conditions favouring them). Further examinations of the historical time series of the ECW cases are therefore needed.
In 2017, we will apply a novel method of using neural networks (NN) to identify the occurrence of extreme convective events in reanalysis data (Ukkonen et al. 2016). Our plan is to utilize this technique to establish a trendline that goes beyond direct observations. This is essentially a downscaling method that takes advantage of the suitability of neural networks to model non-linear phenomena, as well as the consistency and vast sample size of reanalysis data. The NN model would be trained using high-quality observations of e.g. lightning and precipitation that is available for the recent decade or so.

Another approach to explore severe convective weather, adopted in 2016 for large hail (and previously for tornadoes, see Rauhala and Schultz, 2010) is to study which types of synoptic-scale or mesoscale weather patterns typically produce ECW. Objective methods to classify daily weather (circulation) patterns, together with reanalysis data for the past and climate model simulations for the future, could then be used to estimate potential trends in the frequency of ECW. An alternative option is to examine the vertical profiles of temperature, moisture and wind, and the convectivity degree of precipitation reanalysis and climate model data. The problem lays in the small temporal and spatial scales of ECW that cannot be resolved by typical climate models. As a first step, indicators of favorable atmospheric conditions for ECW are examined in EXWE with the aid of the non-hydrostatic convection-permitting HARMONIE weather prediction model.

**Intense sea-effect snowfall.** Coastal snowfall, as a wintertime convective weather phenomena, has been a research topic in EXWE already for a couple of years, however, with rather minor resources. In 2014, a list of past sea-effect snowfall cases prior to the year 2001 was created using manual recordings of weather codes. In order to include the years 2001–2014 into the list, a new identification method was developed in 2015 that utilizes, among others, information in reanalysis data about convectivity of precipitation, i.e., the portion of snowfall as convective. The method was used to select a few sea-effect snowfall cases for further analysis. Simulations of the Merikarvia case in January 2016 with the HARMONIE weather model and verification with weather radar images indicated that the simulation was able to capture the overall situation quite well. In 2017 we aim at better and more thorough understanding of sea-effect snowfall with the aid of state-of-the-art assimilation of remote sensing data in the model simulations. The ultimate goal will be to estimate the frequency and intensity-levels of coastal snowfall near the NPPs.

**Severe freezing conditions.** Freezing rain may fall if precipitation is combined with a rather uncommon thermal vertical profile in the lower atmosphere during wet days. In EXWE in 2015, a freezing detection methodology was developed and has recently been documented in a peer-review scientific paper (Kämäräinen et al. 2017). In that paper, the method for estimating the occurrence of freezing rain in gridded atmospheric datasets was evaluated, calibrated against SYNOP weather station observations, and applied to the ERA-Interim reanalysis for climatological studies of the phenomenon in Europe. In 2016, the freezing detection methodology was refined and was applied to new 6-hourly data from six regional climate models available from the CORDEX initiative. In 2017, the focus will be given to estimates of future changes in probabilities of freezing rain above selected intensity values at the NPP sites on the Finnish coast.

The plans for WP1 in 2017 include three main tasks: 1) Warm-season convective weather phenomena; 2) Detection and characteristics of sea-effect snowfall on the Finnish coast; 3) Severe freezing rain. The partners and person months allocated to WP1 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>FMI</td>
<td>7.8</td>
</tr>
</tbody>
</table>

**2.1.1 Warm-season convective weather phenomena (T1.1)**

The task aims at improved understanding of the atmospheric factors contributing to the formation of warm-season convective weather phenomena. Better understanding of these factors helps in estimating the likelihood and possible intensities of such events in the changing climate at the NPP sites (EXWE in 2018). First, past trends in warm-season convective weather phenomena will be estimated using novel downscaling approaches, neural networks (NN) in particular, together with reanalysis data. The NN model will be trained using high-quality observations of e.g. lightning and precipitation that is available for the recent decade or so. Second, synoptic environments of significant-hail producing thunderstorms and mesoscale factors contributing to the formation and decay of a derecho (syöksyvirtausparvi) will be examined. Simulations conducted by the non-hydrostatic convection-permitting HARMONIE weather prediction model will be analysed to study which mesoscale factors support the bow echo formation, intensification and long duration of the event, and the decay of it. (3.3 person months)
2.1.2 Detection and characteristics of sea-effect snowfall (T1.2)

In order to increase knowledge about sea-effect snowfall on the Finnish coast, the non-hydrostatic convection-permitting mesoscale model HARMONIE model will be used to simulate selected cases. Because HARMONIE can directly resolve physics of the small-scale convective phenomena, no convection parameterization scheme is needed. This is advantageous since parameterization of convection is a large source of errors and uncertainty in lower-resolution mesoscale models. On the other hand, very few in-situ observations are made in the sea area where the convection cells develop. To improve the initial conditions of our simulations we aim to study the impact of assimilation of two types of remote sensing observations on the forecast skill: radiances of a hyper-spectral satellite instrument and weather radar reflectivities. The simulation results give an upper estimate of how reliable model-based assessments can be concerning the occurrence and characteristics of sea-effect snowfall. The aim is to prepare a refereed proceedings journal paper about simulations of the Merikarvia case with conventional data assimilation (i.e., without remote sensing data) and a report about the new simulations. (3 person months)

2.1.3 Severe freezing rain (T1.3)

The preliminary present-day and future probability distributions of freezing rain, presented in EXWE in 2014, will be updated with the new, temporally denser regional climate model data that has been utilized by us in 2016. The methodology to construct future projections of freezing rain above selected intensity values in Europe will be applied at the NPP sites, including analyses of emission scenario uncertainty, climate model-dependent uncertainty, future period-dependent uncertainty and spatial averaging-dependent factors. The applicability of an ice severity index (McManus et al. 2008, Jones, 1998), which describes the overall impact of freezing rain events by taking into account (1) the accreted ice amounts and (2) the effect of wind causing extra stress for ice-loaded structures, will also be tested. (1.5 person months)

2.2 Extreme sea level events (WP2)

The overarching objective of sea level research in EXWE is to assess sea level extremes which are physically possible but so rare that they are not adequately included in current statistics. During SAFIR2018, we intend to gain new advances by complementing existing analyses with high-resolution data which gives a more detailed picture about short-term sea level variations (Fig. 2). We aim to produce probability distributions of extreme sea level events, which are easily applicable in NPP risk assessments.

Figure 2. Several factors with time scales ranging from seconds to centuries affect sea level on the Finnish coast. In the SAFIR2018 programme (red), the focus of sea level research is shifted from long- to short-term sea level phenomena to complement previous analyses in the EXWE project (SAFIR2014, black). However, regular updates to long-term mean-sea-level scenarios are needed when new knowledge accumulates (hence, the dashed red box on left).
In 2017, the advance towards short-term phenomena will continue by including modelling of wind waves as a new research topic in EXWE. Improving our ability to estimate wave risks is a logical continuation to the research concerning the joint effect of sea level and waves, which is a new method of estimating flooding risks more accurately on the Finnish coast (developed in EXWE in 2015–2016). This method will be further improved in 2017 by assessing the conditional probability of sea level variations and wind waves. The third topic concentrates on meteotsunamis and aims to summarize previous research and publish it on a scientific, peer-reviewed forum.

**Joint effect of high waves and high sea level.** During 2015–2016 in EXWE, a new method was developed to calculate a distribution representing the joint effect of water level and wind waves, and to evaluate maximum wave crest height on a steep shore (Leijala et al. 2016, manuscript to be submitted). This method is based on a statistical approach that takes into account local sea level observations and global mean sea level rise, and combines them with a distribution based on local wave height observations. In this method, it is assumed that the variables of sea level and waves are independent.

Although the assumption of the independence of sea level and waves is reasonable, the results at certain locations may be affected by a dependency between the two parameters. This is especially true near the coast, where locations can be sheltered from waves when the wind is blowing from the shore (off-shore winds), while being exposed to waves when the wind is blowing towards the shore (on-shore winds). As short-term sea level changes can depend on the direction of the wind, this can introduce a dependency between waves and sea level variations. It is necessary to examine the conditional probability, which is a measure of the probability of an event given that another event has already occurred.

A general solution for how to treat the conditional behavior in wind waves with respect to sea level is a highly relevant task for improving the new method of evaluating flooding risks on the coast. We estimate that improving the method by studying this fundamental question is more relevant at this stage than applying the existing new method to the NPP sites, which would require local wave measurement campaigns and would only result in a locally satisfactory partial solution. After the current method is improved with the general solution to the issue described above, it can be utilized in different locations along our coastline and will benefit all NPP sites equally in the future by offering improved estimates of flood probabilities.

**Improving wave modelling on the Finnish coast.** Recent developments in assessing the joint effect of sea level variations and wind induced waves have led to the ability to get more reliable information about the combined effect of these two phenomena. The results are also more comprehensive, since the outcome is a probability distribution, not just one single extreme value. However, the new elaborate method sets higher requirements for wave height data. Ideally, decades long observation time series would be used, as is the case for sea level variations, but even the longest wave measurements are short compared to the sea level statistics. Extensive measurements are also mostly conducted in the open sea and do not represent near shore conditions. The heterogeneous nature of the Finnish coastline complicates the matter further, as near shore measurements cannot easily be generalized. Model data is a feasible solution to fill the gap and provide the necessary data for the studies, but there are still challenges concerning the accuracy of the wave model (Björkqvist et al., 2016).

To ensure the quality of the data over a range of different conditions, an extensive model validation is a necessity. FMI has a measurement dataset that covers widely different coastal conditions at over 20 different near shore locations. These measurements have been acquired over the span of five years. This existing extensive dataset gives us an opportunity to perform a massive wave model validation. The validation is not a trivial task, partially because of the high computational demands of the high-resolution wave model.

**Meteotsunamis on the Finnish coast.** Meteotsunamis have been a key area of research within EXWE for several years. As a result, quite a lot of information is now available on this phenomenon, which was previously nearly unknown to occur on the Finnish coast. However, most of the statistics collected exists in various reports and has not been subject to a rigorous scientific peer review. In 2017, the aim is to gather all the available statistics of long-term occurrence of meteotsunamis in Finland and offer it up for review to the scientific community. New analyses with this data will also be conducted to reveal possible connections to climatic patterns.

**Simulating extreme sea levels in the Baltic Sea.** The climate change affects the patterns of low pressure systems, inducing changes in water exchange between the North Sea and the Baltic Sea, and in internal variations of the Baltic Sea level. Studying the effects of these changes on sea level extremes on the coast is important when the safety of coastal infrastructure is assessed. We have previously made numerical sea level simulations using an improved version of a barotropic sea level model based on the Hansen model for the Baltic Sea. It has been used to estimate extreme sea levels at NPP sites in Sweden and Finland, and an article on simulated extreme sea levels based on an 850-year long simulation at Helsinki has been submitted (Särkkä et al., 2016).

The sea level model allows us to study extremely rare but physically plausible sea level events that have not occurred during the century-long observation period. In a task starting in 2017 and continuing in 2018, we aim at assessing the highest sea levels that could occur on the Finnish coast by using synthetic low pressure systems as
an input for the sea level model. In 2017, we will develop the sea level model further and assess its reliability in simulating the past extreme sea level events on the Finnish coast. This will form a basis for further studies in 2018 of different low pressure tracks and depths and their effect on sea level extremes.

In 2017, WP2 is planned to consist of four tasks (T2.1-T2.4). The goals and research contents of the tasks are described in the following. Partners and person months allocated to WP2 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
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</tr>
</tbody>
</table>

2.2.1 Conditional probability of sea level variations and wind waves (T2.1)

The method to estimate the joint effect of high sea level and high wind waves will be further improved by studying the dependency between the two parameters. Previous studies have assumed that the wave and sea level distributions are independent of each other, which is not necessarily the case. In the future, this method will replace the current method of estimating the probabilities of flooding heights on the Finnish coast. The improved method will be documented and the findings reported in a scientific, peer-reviewed publication. The work will be supported by a 2-month research visit to NERSC (Nansen Environmental and Remote Sensing Center) in Bergen, Norway. (3.5 person months)

2.2.2 Validating a high-resolution wave model for the Finnish coast (T2.2)

The task will provide an extensive observation based validation of a high-resolution wave model on the Finnish coast. The validation study will provide valuable information about the performance of the wave model in different geographical and atmospheric conditions. The possibility to use a model with a slightly coarser resolution will also be studied, because this would enable us to produce longer time series with reduced computational cost. The results of the validation and other findings will be published in scientific peer-reviewed journals. (3 person months)

2.2.3 Scientific publication of long-term meteotsunami statistics (T2.3)

This task will gather all the previously obtained statistics of meteotsunami occurrence in the Gulf of Finland. The statistics obtained from the archived tide gauge records (1920s–1980s for the Hanko and Hamina tide gauges) will be complemented with the modern digital observation data with 1-min resolution (available from all tide gauges from 2004, studied in EXWE in 2016). This will result in a unique time series spanning nearly a century. The correlation of this time series with climate variables such as the NAO index (North Atlantic Oscillation) will be studied, possibly giving new insights on why the occurrence of meteotsunamis has decadal variability on the Finnish coast. The results will be published in a scientific peer-reviewed journal. (2 person months)

2.2.4 Simulating extreme sea levels in the Baltic Sea (T2.4)

This task will estimate numerically Baltic Sea levels in the 20th century and develop methods to assess future sea level extremes on the Baltic coast. We perform a hindcast simulation of Baltic Sea levels in 1900–2010 by using ERA-20C reanalysis data as atmospheric forcing. The hindcast simulation enables model validation with sea level observations from Finland and Sweden. Based on the simulation results, we perform case studies of highest sea level extremes in the past. The results of the hindcast simulation will be presented in a scientific peer-reviewed paper to be submitted in 2018; a draft is presented for the reference group in 2017. (1 person month)

2.3 Atmospheric Dispersion Tool (WP3)

The goal of this workpackage is to develop an integrated dispersion and dose assessment toolset by connecting the SILAM dispersion model with state-of-the-art dose-assessment software. Moreover, the dispersion calculations are supported with the HARMONIE high-resolution meteorological model simulations.

In 2016, updates of the HARMONIE high-resolution meteorological model have been carried out and specific new simulations by HARMONIE for EXWE are being analysed. HARMONIE was set up on a domain covering southern Finland and adjacent seas with a grid having a step-ping of 500 m in the horizontal. Hind casts were produced for April and May 2015, when a rich set of remote-sense measurements by SODAR and LIDAR, as well
as in situ data from a tower were available at the Loviisa nuclear power plant (Jurvanen, 2015). Validation of the
hind-casts using these data is on-going.

This work in 2017-2018 consists of evaluation of the meteorological modelling system of further development,
integration and application of the dispersion modelling system SILAM.

The final aim is to develop an integrated dispersion and dose assessment tool relying on properly evaluated
meteorological modelling. Based on previous discussions with parallel projects in SAFIR2018 (PRAMEA, CASA),
co-operation will be made in 2017-2018 with VTT for this work.

The following steps will be taken:

- Evaluation of high-resolution meteorological modelling, especially concentrating on studying the ther-
  mal internal boundary layer (TIBL)
- Integration of local-scale dispersion (plume/Lagrangian) model to SILAM to cover local dispersion and
  coastal meteorological effects properly at resolutions less than 1 km (2017–2018). (FMI own resources)
- Final evaluation and assessment of the integrated system together with VTT (2018).

In 2017, WP3 consists of two tasks. The goals and research contents of the tasks are described in the following.
Partners and person months allocated to WP3 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>FMI</td>
<td>2.5</td>
</tr>
</tbody>
</table>

2.3.1 Evaluation of high-resolution meteorological modelling (T3.1)

In 2016, the operationally used HARMONIE grid spacing of 2.5 km has been reduced to 0.5 km, and for that pur-
pose, topographic data files (surface type, surface elevation, land use, etc) ingested by HARMONIE have been
prepared. In 2017-2018, hind-casts performed by the high-resolution model will be carried out and validated by on-
site measurements, and used in an assessment of how dispersion is affected by flow features related to the con-
trasting roughness and thermal inertia of land and sea surfaces.

In 2017, we will specifically study the formation, strength and occurrence of thermal internal boundary layer
(TIBL) in Finland and the effect of TIBL on coastal mixing of the emissions at the NPP sites. The TIBL layer is
formed when a stable marine air mass is transported to a warm land surface during the day; the bottom portion of
the marine air mass becomes unstable due to the heat gained from the warm land surface. This layer is important
because it controls the vertical mixing in a coastal region. The TIBL height grows with the distance from the shore-
line, and when reaching the height of the plume, the pollutants can be mixed down to the surface. (2.0 person
months)

2.3.2 Meteorological model integration to SILAM (T3.2)

Integration of local-scale dispersion (plume/Lagrangian) model to SILAM to cover local dispersion and coastal
meteorological effects properly at resolutions less than 1 km (0.5 person months)
## 3. Deliverables 2017

<table>
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<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
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<tr>
<td>D1.1.1</td>
<td>A report on past trends in warm-season convective weather phenomena using novel downscaling approaches</td>
<td>2</td>
<td>30.11.2017</td>
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<td>Description: a report documenting the use of alternative approaches to examine occurrence, characteristics and trends in extreme convective weather. Approval: the report together with summarizing ppt-slides sent to RG2 for comments.</td>
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<td>D1.1.2</td>
<td>A conference abstract and presentation about synoptic environments of significant-hail producing thunderstorms in Finland.</td>
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<td>D1.1.3</td>
<td>A report on mesoscale factors contributing to derecho formation and decay</td>
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<td>Description: a case-study report presenting results from HARMONIE model simulations of a derecho in Northeastern Europe. Approval: the report together with summarizing ppt-slides sent to RG2 for comments.</td>
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<td>A report about sea-effect snowfall simulations</td>
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<td>D1.3.1</td>
<td>A report on the present-day and future annual probabilities of severe freezing rain at the NPP sites.</td>
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<td>Description: A report presenting frequency curves for Olkiluoto, Lovisa and Pyhäjoki sites, with uncertainty estimates, considering differences between models, emission scenarios and geographical aspects. Approval: The report together with summarizing ppt-slides sent to RG2 for comments.</td>
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<td></td>
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<td></td>
<td>3 31.12.2017</td>
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<table>
<thead>
<tr>
<th>D2.3.1</th>
<th>Scientific paper summarizing 100 years of meteotsunami statistics on the Finnish coast</th>
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<tr>
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<td>Description: Peer-reviewed scientific publication on long-term variation of meteotsunami occurrence in the Gulf of Finland and its correlation with climatic variables. Approval: The manuscript has been submitted to a scientific journal.</td>
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<table>
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<tr>
<td></td>
<td>Description: Draft of a scientific publication on the sea level hindcast simulation. Approval: The draft of the manuscript has been presented to the reference group.</td>
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<tr>
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<th>Report documenting the developments in high-resolution meteorological modelling with HARMONIE</th>
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<th>D3.1.2</th>
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<th>D3.1.3</th>
<th>Technical report describing the integration of the SILAM dispersion model with the high-resolution NWP-model HARMONIE</th>
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<td>0.5 31.12.2017</td>
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</tbody>
</table>

| Total pm | 19.8 |
4. Project organisation

The project manager is Dr Kirsti Jylhä, Finnish Meteorological Institute (FMI), Climate Service Centre (IKE). FMI is responsible for the whole project. The FMI research unit IKE is responsible for WP1, the Marine Research unit (MER) for WP2, and the Atmospheric Composition Research unit (IKO) for WP3. Related to WP3, discussions with VTT will continue in 2017 through the parallel projects in SAFIR2018 (PRAMEA, CASA).

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kirsti Jylhä</td>
<td>Senior research scientist</td>
<td>FMI (IKE)</td>
<td>Project manager, WP1</td>
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<tr>
<td>Antti Mäkelä</td>
<td>Group leader</td>
<td>FMI (IKE)</td>
<td>Deputy project manager, T1.1</td>
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<tr>
<td>Andrea Vajda</td>
<td>Group leader</td>
<td>FMI (IKE)</td>
<td>T1.3</td>
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<td>Terhi Laurila</td>
<td>Research scientist</td>
<td>FMI (IKE)</td>
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<td>Taru Olsson</td>
<td>Research scientist, PhD student</td>
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<td>Matti Kämäräinen</td>
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<td>Otto Hyvärinen</td>
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<tr>
<td>Anna Luomaranta</td>
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<tr>
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<td>Research scientist (PhD)</td>
<td>FMI (MER)</td>
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<tr>
<td>Jan-Victor Björkqvist</td>
<td>Research scientist, PhD student</td>
<td>FMI (MER)</td>
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<tr>
<td>Jani Särkkä</td>
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<td>FMI (MER)</td>
<td>T2.2, T2.4</td>
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<tr>
<td>Ari Karpinnen</td>
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<td>FMI (IKO)</td>
<td>T3.1 - T3.2</td>
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<tr>
<td>Carl Fortelius</td>
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<td>Mikhail Sofiev</td>
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<td><strong>Total</strong></td>
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</table>
5. Risk management

A potential risk may be related to archives and stores of observed past cases and model data. Technical problems such as computer disk crashes will be minimized by double archival. A challenge is to optimise the use of data and computation time. By using the already established connections to data servers in Europe it is not necessary to, for example, download the data locally to FMI, but to run the data analysis on remote servers. Besides, the STORNEXT backup system is available at FMI in case of urgent and large data storage volumes.

Other risks include our capability to get the necessary reliable information (extreme events occurrence and related characteristics: origin, intensity, location, duration) from non-observation data. Also, whether we are able to get a representative number of events for statistical analyses poses a risk.

The above-mentioned risks can be managed by cross-validation of the developed ingredient based approaches and pattern recognition techniques with known historical cases. Choosing to use reanalysis data in addition to in-situ observations bring us the benefit of mapping the ingredients that are also possible to model. Thus, we can iterate the methods to make them more accurate. This iteration will improve the methods and support our work when making projections about climate change impacts on extreme weather events.
References


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### The Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2018)

#### Resource Plan for 2017

**EXWE**

Extreme weather and nuclear power plants

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Volume</th>
<th>Personnel</th>
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<td><strong>WP1 - Extreme weather</strong></td>
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<tr>
<td><strong>WP2 - Extreme sea level events</strong></td>
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**Comments:**

Memb fee: Please explain the membership fees in international projects included in the cost budget (VYR funding 100% if approved).

Describe possible in-kind work contribution here (organisation and person months, use of equipment etc.).

Other explanatory comments.
SAFIR2018 Project plan

GENXFIN
Safety of new reactor technologies

Sami Penttilä, Jarno Kolehmainen, Mikko Ilvonen, Ville Tulkki
VTT Technical Research Centre of Finland Ltd

Karoliina Ekström, Antti Rantakaulio
Fortum Power and Heat Oy
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1. Research theme and motivation

The mission of the GENXFIN project is to improve scientific and technologic expertise in the field of advanced nuclear energy technologies and related processes through national and international collaboration. The main objective of the GENXFIN project is to increase knowledge on safety issues of selected advanced reactor concepts e.g. in terms of licensing requirements and material challenges. In addition, the purpose is to coordinate participation in various international forums and working groups as well as support activities related to the new national doctoral programme network. The motivation for the project derives from national strategy as described in the Ministry of Economic Affairs and Employment (MEAE) reports [1, 2] highlighting new reactor technologies which will be an important part of future research activities in Finland. Only by being part of the international development, can Finland have an impact, utilising the high Finnish safety culture. This is only possible through separate project supported by research organizations and other domestic key players in the field.

While common understanding is that the advanced nuclear designs will take a long time to be adopted, there are several developments which may indicate that the next generation of nuclear power is deployed faster than previously anticipated. 800 MWe sodium cooled reactor achieved full power in Russia early 2016 with contracts of several to be built in China. China is constructing a dual unit gas cooled pebble bed reactor with estimated grid connection in 2017. In USA, NuScale is in the process of licencing a multi-reactor SMR (Small Modular Reactor) to be built in Idaho and a designer of integral molten salt reactor is applying for a one billion dollar loan guarantee for licensing and building a prototype reactor. Driver for these changes is the climate change and the need for the zero-carbon energy system as well as the fact that the innovative reactors provide alternatives to various uses of fossil fuels also in fields other than pure electricity production. In order to assess both the safety significance as well as the Finnish prospects regarding this new generation of nuclear power the national actors need to be informed. GENXFIN serves this need in the national framework. Also, in order to better take advantage of the future potential for nuclear power the needs and areas of focus of Finnish nuclear research must be specified, and a long-term strategy for the development of research activities needs to be drafted.

According to SET-plan (The European Strategic Energy Technology Plan), innovative nuclear energy technologies will be an important part of future development towards zero carbon society. Also the national research strategy for the nuclear energy through 2030 (YES) [1] recognises the need for future reactor technology R&D and following its recommendations GENXFIN was initiated in 2016. For those purposes GENXFIN has emphasized the study on different reactor technologies (SMRs, Gen4, fusion) on a yearly basis that are recognised the most important nationally. Research activities are needed for future reactor technologies both nationally and internationally and this is coordinated by GENXFIN project.
1.1 Background and state-of-the-art

In 2004, a networking project on future reactor technology topic was initiated by MEAE (Ministry of Economic Affairs and Employment). All major nuclear field players including research organizations, universities, power companies and safety authority were invited to participate in the project. For the past ten years, this networking project called GEN4FIN (http://gen4fin.vtt.fi) has been active in keeping abreast of Generation IV (Gen4) reactor technology. As a new topic, SMR was included first time in the research agenda in 2012. As many of the research issues are cross-cutting across reactor types and generations, the best way to disseminate the information collected in this effort is the SAFIR2018 programme.

In the long term, the goal in GENXFIN is to create new expertise on nuclear energy and business opportunities for the Finnish industry by promoting technology transfer, innovative industrial processes and materials technology. The GEN4FIN network created a national roadmap on Generation IV [3] where the Finnish participation of Gen4 research and development was planned. The network participated actively in the national research strategy project (YES) under the MEAE in 2013 - 2014. In 2015, the GEN4FIN network continued its activities in the traditional mode of operation and made plans for the coming years in the current operating environment that has changed since the start. This planning phase led to a decision to initiate the GENXFIN in 2016. As a result, a “Vision paper” on advanced reactor technologies R&D in national level was completed in 2016. This work was done under the guidance of GENXFIN ad hoc steering group.

Nowadays research activities on SMRs have increased especially in UK, China, Korea, Russia and USA [4-12]. The main reason for their attractiveness is that they can be used for multipurpose applications of desalination, small scale power generation and district heating. SMRs offer a viable alternative to large present-day LWRs. SMRs fulfil many of the needs and aspirations related to flexibility and manageable capital investments in present economic situation. SMRs are also designed to be especially safe and many SMR designs include numerous passive safety features. SMRs can be seen attractive also in countries where the electricity grid is not capable for large reactors and where is difficult to provide power in distant regions. There are a lot of potential customers for SMRs. China is aiming to have 110 operational nuclear reactors by 2030 of a capacity of 88 GW meaning 6 to 8 nuclear reactors every year from 2016 for the next five years [4]. NuScale is aiming to build its first SMR plant in the USA by 2023, and believes it could build its first UK plant by the mid-2020s. Last November, the UK government announced plans to invest at least £250 million ($352 million) over the next five years in an “ambitious” nuclear research and development program to include a competition to identify the best value SMR design for the UK. In March 2016, the Department of Energy and Climate Change (DECC) officially launched the first phase of the competition by publishing a request for expressions of interest. Globally, the National Nuclear Laboratory considers the potential market for SMRs of all kinds to be up to 85 GWe by 2035 [5]. In addition, the construction of CAREM in Argentina, the first natural circulation integral pressurized water reactor is on schedule with first criticality aimed by 2018 [6].

USA is emphasising traditional water-cooled based SMRs primarily because the NRC (US Nuclear Regulatory Commission) has the most experience with licensing large LWRs [7]. On the other hand several USA based startups aim to develop SMRs based on the alternative coolants and moderators [7]. Many of the SMR designs in the USA are aiming to directly replace aging coal power plants. On the other hand, many countries seem to be more open to innovative designs, e.g. Russia to lead-cooled fast reactors and floating power plants, India to thorium based heavy water reactors, Canada to heavy water SCWR / small scale SCWR concepts, and China to high temperature gas-cooled reactors and their willingness also to proceed with licensing procedures faster than in other countries. Also worth to mention is very ambitious Terrapower’s Travelling Wave Reactor (TWR, Sodium cooled), as well as Thorcon Power (Molten salt reactor) project. They both are pushing forward
demo reactors which should be available in mid-2020s and in 2020, respectively [8, 9]. Consequently, many nations worldwide, like Argentina, Canada, China, France, India, Japan, Russian Federation, South Africa and United States, are developing their own innovative technologies in this field, using various types of concepts.

As indicated above, SMRs are under development work in different countries and as such the EU project proposals TREND-SMR (Technical Requirements for SMR, coordinated by VTT) and E-SMART (European Small Modular supercritical Water Reactor Technology) were submitted in October 2016 to Euratom Call. Both contribute to the activities of future type of reactors through Generation IV International Forum (GIF) and also to NUGENIA since mid-2015 this international association for R&D on Gen II & III reactors made a decision to support the SCWR concept via its Technical Area 6 (TA6) “Innovative LWR designs and technologies”. By studying these two reactor concepts the GENXFIN project is very well in-line also with the activities of NUGENIA TA6.

Research on future reactor technologies has an educational role in Finland but it is also a platform for technology development. Licensing of innovative reactor concepts like SMRs is interesting from a national perspective. It is important to consider the feasibility of new technologies including scientific, technical, economic and political aspects.
1.2 Objectives and expected results

The Finnish actors have achieved a significant role in performing and directing scientific research and technological development for nuclear reactor concepts in global forums. The main objective of the GENXFIN project is to improve scientific and technologic expertise in the field of innovative nuclear energy technologies. The knowledge is needed nationally to enable future nuclear reactors being deployed in a reasonably near future in Finland. Also international collaboration with IAEA INPRO (International Project on Innovative Nuclear Reactors and fuel cycles), OECD/NEA CSNI (Economic Co-operation and Development / Nuclear Energy Agency the Committee on the Safety of Nuclear Installations) and other global forums such as GIF, is needed. The mission is not only to create enabling knowledge pool in Finnish needs, but also enable new business activities for the Finnish industry through enhanced technology transfer, innovative process development, and materials engineering. Also the safety authorities benefit from the outcomes of this project since they have the possibility to follow and steer the development work of the future reactor technologies. The activities in the programme will cover scientific, technological and industrial goals. The potential of new technologies on the national level is evaluated and the Nordic research on this theme is followed through NKS (Nordic Nuclear Safety Research) and Energiforsk. An integral part of the project is to enhance collaboration between research organisations and power companies that define the Finnish research focus. Research & education organisations, safety authorities, manufacturing industry and power companies as well as ministries and other associated organisations are following outcomes of the project e.g. through travel reports and ad hoc meetings. The updated GEN4FIN webpages will be used for document sharing which enhances discussion on the topic.

GENXFIN will follow developments and coordinate national projects related with new reactor technologies, focusing mainly on SMRs but including also Gen4 and CHP (Combined Heat & Power) topics. Additionally, combined technical-economic issues will be covered through case studies in the coming years. Load following and grid development related topics with new kinds of electricity generating systems as well as advanced fuel cycles will be followed within the project. The objective of the GEN4FIN project has been to enhance national expertise in science and technology of nuclear reactors. The project has had international cooperation within GIF, EERA (European Energy Research Alliance), IAEA (International Atomic Energy Agency) and ESNII (European Sustainable Nuclear Industrial Initiative). This type of international cooperation is meant to continue within GENXFIN project in 2017 focusing mainly on the different IAEA activities (through consultant and technical meetings) as well as INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles), and also WNA (World Nuclear Association) Working Group CORDEL (Cooperation in Reactor Design Evaluation and Licensing) and GIF SCWR (Supercritical Water Reactor) M&C (Materials & Chemistry) Project Management Board (PMB).

Licensing of SMRs is interesting from a national perspective as well as internationally [5-7, 10]. Especially after Fukushima accident there has been considerable discussion about safety issues. SMR designs with power levels of less than 300 MWe are being developed in several countries nowadays (USA, Russia, China, Korea etc). Those designs benefit from enhanced safety, flexibility and reliability and longer fuel cycles [11]. While there are several potential advantages with these reactors, they are also confronted with multiple challenges. Important among these challenges is to have these new reactor designs licensed by national regulatory bodies. Because of the many novel features incorporated in different SMR designs, careful and thorough licensing procedures are critical to maintaining safety of SMRs.
This project is based on assessment of technical features of SMRs in terms of licensing requirements including severe accidents. Major gaps in the knowledge will be identified in licencing issues e.g. through system engineering approach and in severe accident management.

The most challenging cases are expected to be the available plant descriptions in literature which are not enough detailed in order to benefit this work directly. More information is needed on dependencies of different SMR DiD (Defence-in-Depth) concepts and especially separation of DiD levels, but this will be evaluated through the study in 2017. The Western European Nuclear Regulators Association DiD levels are presented in Table 1. NuScale will submit their design certification application by the end of 2016 to NRC which might give some insight on the above mentioned aspects on this specific reactor design.

Table 1. WENRA (Western European Nuclear Regulators Association) Defence in Depth levels.

<table>
<thead>
<tr>
<th>Levels of defence in depth</th>
<th>Objective</th>
<th>Essential means</th>
<th>Radiological consequences</th>
<th>Associated plant condition categories</th>
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</thead>
<tbody>
<tr>
<td>Level 1</td>
<td>Prevention of abnormal operation and failures</td>
<td>Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits</td>
<td>No off-site radiological impact (bounded by regulatory operating limits for discharge)</td>
<td>Normal operation</td>
</tr>
<tr>
<td>Level 2</td>
<td>Control of abnormal operation and failures</td>
<td>Control and limiting systems and other surveillance features</td>
<td>No off-site radiological impact or only minor radiological impact</td>
<td>Anticipated operational occurrences</td>
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<tr>
<td>Level 3</td>
<td>3.a Control of accident to limit radiological releases and prevent escalation to core melt conditions</td>
<td>Reactor protection system, safety system, accident procedures</td>
<td>Postulated single initiating events</td>
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<td></td>
<td>3.b Control of accidents with core melt to limit off-site releases</td>
<td>Additional safety features, accident procedures</td>
<td>Postulated multiple failure events</td>
<td></td>
</tr>
<tr>
<td>Level 4</td>
<td>Control of accidents with core melt to limit off-site releases</td>
<td>Complementary safety features to mitigate core melt, Management of accidents with core melt (severe accidents)</td>
<td>Off-site radiological impact may imply limited protective measures in area and time</td>
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<tr>
<td>Level 5</td>
<td>Mitigation of radiological consequences of significant releases of radioactive material</td>
<td>Off-site emergency response, Intervention levels</td>
<td>Off-site radiological impact necessitating protective measures</td>
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The expected results of the project can be listed as following:

a) Assessment of licensing requirements in selected SMR designs beyond those of GENXFIN 2016.

b) Assessment of selected SMR plant systems for management of serious accidents. Numerical assessment of off-site radiation doses due to some hypothetical atmospheric radioactive releases from SMR plants.

c) Information dissemination for the national network through travel reports and GENXFIN ad hoc meetings when necessary (relevant forums & workshops, IAEA, WNA WG CORDEL and GIF activities).
1.3 Exploitation of the results

The project results will assist the safety authorities to prepare for possible future applications on new reactor concepts. They can also be exploited by possible coming licensees. Further, the results can be utilised in development of current codes, as application of these on new systems may reveal possibilities for improvement. New innovative methods will increase knowledge in this topic and shall be used in analysing the overall safety of new reactor concepts. In Finland, no new-build plant is acceptable without a feasible strategy for managing severe accidents (STUK Ydinturvallisuusohjeet (YVL) 2.2). The studies performed in this project further strengthen possibilities for developing simulation and analytical capabilities on safety.

GENXFIN will maintain strong links and will interact with other organisations performing R&D on the innovative reactor technologies especially in IAEA and GIF. Results of the project will be presented at relevant conferences, workshops and symposiums, e.g. IAEA Technical Meetings.

A major part of the project is information dissemination that involves all national stakeholders. This will be executed through updated GEN4FIN webpages (http://gen4fin.vtt.fi/). VTT will set up a platform for the exchange of project deliverables (final and draft versions), important reference documents and other documents relevant for the project. This will be done by using updated GEN4FIN webpages where GENXFIN members can share information and resources.

In practice, Gen4 reactors are seen as a long term objective, however, several innovative reactor types are being deployed faster than previously anticipated. Meanwhile SMRs are seen technically and economically feasible even earlier. Many of the new developments for the new reactor technologies can be applied in current reactors and those under construction within 5 years.
1.4 Appropriateness of the project to SAFIR2018 programme

In the SAFIR2018 framework plan it is mentioned that “the focus areas of the nuclear energy research must be assembled into broad national programmes”. This includes also future nuclear energy research which is a good platform for educating a new scientist in the field. This SAFIR2018 project proposal GENXFIN is forming the new way of grouping the national efforts of nuclear energy research activities in Finland.

The GENXFIN project is well suited to SAFIR2018 programme. The safety of current and near-future reactors requires profound training and commitment. This can only be achieved in long-term research where new experts are educated and the safety is challenged with cross-disciplinary thinking. New reactor systems, including innovative LWR technologies like SMRs and SCWRs, provide a platform for this kind of activities. In particular the framework plan cites that “the R&D work on SMRs has been commenced within international co-operation e.g. through IAEA INPRO and during the SAFIR2018 programme it may also be appropriate to conduct a suitable study of SMRs”. The SAFIR2018 programme emphasises also international collaboration which is the key element in this project. This project proposal fits under the research topic of “Plant safety and systems engineering”.

All results of GENXFIN project are generic and can be applied directly on the existing LWRs and new innovative reactor designs. Therefore the outcomes of this project will contribute to the continuous improvement of nuclear safety of the existing NPPs as well as the optimization of the safety characteristics in the design of future reactors. Working on the topic of SMRs provides an excellent opportunity to investigate innovative features of safety and security. This kind of topics support also public acceptance of nuclear technology.
1.5 Education of experts

The project promotes participation in international research networks in Europe and globally. In this manner Finnish researchers and their support groups can contribute to overall development of nuclear safety. This project supports especially young scientists who are working with their dissertations in the way of helping their international networking. For these purposes, this project supports participation of scientist in the relevant conferences and symposiums. Dissemination of results in international seminars and technical forums are seen as a key objective in order to educate new scientists in the global nuclear field. Also strong collaboration with other partners involved with advanced reactor systems will lead to improvement in understanding of key phenomena, advancement in technology and completion of different innovative concepts.
2. Work plan

General description

This project proposal consists of three work packages: WP1 Safety features of SMRs, WP2 International cooperation and WP3 Project management.

In 2016, literature review was made on potential issues regarding the adoption of SMR designs in terms of licensing and material challenges. The major issue in both tasks was the lack of public data. It is evident that more information is needed on different defence in depth concepts and especially about the separation of different defence in depth levels. A joint report was prepared together with Fortum (in-kind) on this topic. With respect to the materials in general, a limited amount of information was available in public sources. However, data on Korean SMART design was available including RPV and internals. Also some joining and manufacturing issues and NDT inspection cases were discussed in 2016.

In 2017, WP1 is focusing especially on NuScale design. The design certification application of NuScale is expected to be available in 2017, since it should have been submitted to NRC by the end of 2016. The potential critical issues with regards to the selected SMR designs are investigated through licencing challenges. This project examines how Finnish guidelines have engaged in the process of licensing new reactor designs, and demonstrates both similarities and differences between selected countries. In many cases, designers have emphasized the safer design and deployment features of SMRs and attempted to use those features as reasons to get existing licensing requirements diluted. This raises the concern that the promised safety enhancements in SMR designs could be offset by a simultaneous relaxation of licensing requirements. The goal is however, to achieve at least the same level of safety as the current new plants that are being built. A challenge is how this can be achieved with current requirements that are very specific and demanding and which are based very strongly on current reactor concepts.

Licensing requirements are quite different in Europe and the USA. Since the WENRA requirements are at quite high level, there are not many conflicts, but challenges appear when interpreting these requirements for each country. More differences exist between the USA and Finland, since YVL guides are more prescriptive than WENRA requirements. The work has started with the evaluation of STUK’s regulations in 2016, which came into force from the beginning of 2016. Further, the systems engineering approach will be taking into account when assessing the Finnish licensing approach in 2017.

In 2016, the objective of WP2 was the information dissemination on EU and other collaboration projects to the national network focusing on IAEA, ESNII task force (European Sustainable Nuclear Industrial Initiative), EERA JPNM (European Energy Research Alliance Joint Program Nuclear Materials) as well as GIF SCWR (Supercritical Water Reactor) activities. For this purpose, an ad hoc GENXFIN Steering Committee (SC) was established in 2016 and two meetings were held in January and May. Active participation in EERA JPNM and ESNII gave possibilities to affect the Horizon2020 call text and become a partner in EU projects. The European
energy strategy is strongly based on future reactors. It is important to participate in the development in the early phase as joining later is more difficult.

For 2017, the objective of WP2 is to continue the collaboration with international activities and participate in relevant networks and increase the interdisciplinary research activities and knowledge transfer within Finland in advanced reactor technology areas. IAEA activities e.g. through consultant & technical meetings as well as INPRO and CRPs (Coordinated Research Projects) are the main scene for this and participation in WNA Working Group CORDEL will create new knowledge on the licencing issues of future nuclear energy systems. Also participation in the GIF will enable VTT and other stakeholders in Finland to remain in active role in the global nuclear field and also create new knowledge in Finland also in research areas outside the core nuclear area, e.g. innovative material solutions. GIF maintains a long-standing and collaborative relationship with the IAEA INPRO and cooperation has been ongoing for several years. R&D activities within GIF are carried out at the project level e.g. focusing on material challenges and thermal-hydraulics and involve all sectors of research community. The activities within the NUGENIA (Nuclear Generation II & III Association) association concerning SMRs and other innovative LWR technologies like SCWRs will be followed as well.

In WP3 several different near term goals were set in GENXFIN “Vison paper” in 2016. This document pointed out the need for increase the impact of Finnish research work performed on advanced reactor concepts. Also the need for increasing the size of Finnish stakeholder group interested in advanced reactor concepts and the Finnish funding potential related to advanced reactor concepts was emphasised.

In 2017, the third work package WP3 deals with the coordination of the project. Information dissemination is an important part of this project which will be executed through travel reports and new GENXFIN webpages. Also GENXFIN ad hoc meetings will be arranged when necessary.

Table 2. The national and international cooperation within GENXFIN.

<table>
<thead>
<tr>
<th>Work package</th>
<th>Cooperation/travelling</th>
<th>Deliverable type</th>
<th>Who</th>
</tr>
</thead>
<tbody>
<tr>
<td>1, 2</td>
<td>IAEA INPRO</td>
<td>Travel report</td>
<td>Fortum / VTT</td>
</tr>
<tr>
<td>1, 2</td>
<td>WNA WG CORDEL</td>
<td>Travel report, in-kind</td>
<td>Fortum</td>
</tr>
<tr>
<td>2</td>
<td>GIF SCWR M&amp;C PMB</td>
<td>Travel report</td>
<td>VTT</td>
</tr>
</tbody>
</table>

12
2.1 Work package 1 (WP1) Safety features of SMRs

This work package investigates the potential issues regarding the adoption of SMR designs in Finland. The WP1 focuses on the licensing barriers inherent in the SMRs e.g. emphasising accident management in a regulatory environment created for large LWRs. The SAMG (Severe Accident Management Guidelines) are taken into account when applicable.

Table 3. Partners and person months allocated to WP1.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fortum (in-kind, separate internal funding)</td>
<td>2.0</td>
</tr>
<tr>
<td>VTT</td>
<td>3.0</td>
</tr>
<tr>
<td>Total</td>
<td>3.0</td>
</tr>
</tbody>
</table>
2.1.1 Task 1 (T1.1) Licensing

The objective of this task is to continue with the work done in 2016 in order to evaluate STUKS’ regulations (e.g. YVL B.1 guidelines) against passive safety features in SMRs. Also other YVL guides and nuclear legislation will be evaluated and included in the evaluation. When these requirements were indicated and evaluated in 2016, their safety significance is assessed in 2017 e.g. by using systems engineering approach. Also different options are studied to enable SMR licensing, should the requirements be modified for SMRs or could an adequate level of safety be ensured otherwise. In addition, severe accident in licensing of SMRs is considered also taking into account Severe Accident Management Guidelines (SAMG). Some assumed radioactive releases (due to selected accidents) possibly by affecting multiple units will be assessed with respect to their radiological consequences in the environment. The results of this assessment will be checked against appropriate YVL Guides.

In 2017 the work is continued using systems engineering approach as a basis for the evaluation. This way system engineering principles are applied in Finnish licensing approach. Systems engineering approach enables technology independent comparison against YVL guides. One or two reactor concepts are selected and a case study is made. Through this case study it can be evaluated how suitable systems engineering approach can be for the licensing activities.

In Finland, STUK requires the consideration of severe accidents in the licensing of new reactors. This is the trend increasingly in other countries as well. The YVL Guides contain requirements for the management of severe accidents and also for mitigation of their radiological consequences.

In most prominent SMR designs, molten corium is managed by in-vessel melt retention with heat transfer through the RPV wall. Combustible mixtures with hydrogen are in most cases prevented by the use of recombiners. Removal of residual decay heat from the containment relies in most cases on passive heat transfer systems. Many designs have several barriers to prevent the transport of fission products out of the RPV and containment. Other severe accident phenomena to be considered include the possibility of re-criticality and steam explosions.

Elaborating further the work done in 2016, these systems in various SMR designs are checked against the requirements in the appropriate YVL Guides, e.g. B.1 (Safety design), B.3 (Deterministic safety analyses), B.5 (Reactor coolant circuit), B.6 (Containment) and B.7 (Internal and external hazards). Conclusions will be given on the suitability of the YVL Guides B.* for SMR licensing.

Possible implications of the SMR designs on severe accident management guidelines (SAMG) will be also considered.

Finally, some assumed radioactive releases, due to selected accidents (possibly affecting multiple units), will be assessed with respect to their radiological consequences in the environment. The results of this assessment will be checked against appropriate YVL Guides, e.g. C.4 (Assessment of radiation doses to the public), C.5 (Emergency arrangements) and C.7 (Radiological monitoring of the environment). Conclusions will be given on the suitability of the YVL Guides C.* for SMR licensing.

In general, the purpose of this task is to identify current guidelines which are in conflict with SMR type reactors and evaluate how guidelines should be improved.
2.2 Work package 2 (WP2) International co-operation

The main objective of WP2 is to coordinate participation in international forums and working groups.

The goal is to participate in research for international networks and increase the interdisciplinary research activities and knowledge transfer within Finland in future reactor technology areas. On the European level focus is on different IAEA activities like consultant & technical meetings and e.g. in INPRO which is the main scene for innovative nuclear reactors and fuel cycles R&D. Participation in GIF activities will create new knowledge for Finnish stakeholders especially in terms of novel materials solutions and manufacturing processes. Partnership in different international working groups and forums has been identified as one main tool in VTT’s international strategy and e.g. participation in the NUGENIA activities has enabled Finland to remain in active role in the European nuclear field. This includes e.g. materials and manufacturing technology, severe accident management as well as development and validation of system codes (e.g. APROS). Activities of the NUGENIA association in the technical area 6 will be followed in terms of SMR and SCWR concepts as well as Halden MTO program on “Advanced reactors”. In addition, a consortium which was formed (to develop a fleet of 7 GWe of SMRs) in October 2016 by Rolls-Royce to launch SMR in the UK [12] will be followed via possible workshops and public sources. This project will be the forum for information exchange from the relevant working groups, conferences and symposiums on advanced reactor technologies through travel reports and possible ad hoc meetings.

Table 4. Partners and person months allocated to WP2.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.8</td>
</tr>
<tr>
<td>Total</td>
<td>1.8</td>
</tr>
</tbody>
</table>
2.2.1 Task 1 (T2.1) International networking

The main goal of this task is to improve scientific and technological expertise in the field of innovative nuclear energy technologies and related processes through collaboration with relevant working groups and forums.

IAEA INPRO (International Project on Innovative Nuclear Reactors and fuel cycles) is a forum for nuclear technology holders and users to consider jointly international and national actions that would result in required innovations in nuclear reactors, fuel cycles or institutional approaches. INPRO Collaborative Project “Case Study for the Deployment of a Factory Fuelled SMR” is a project, the purpose of which is to examine, in details, legal and institutional issues for export deployment of a transportable nuclear power plant (TNPP) with a factory fuelled and tested reactor and to investigate other aspects of transportable and modular reactor facilities. The current participating countries are Armenia, Canada, Finland, France, Indonesia, Romania, Russia and USA. So far four Consultants’ Meetings in 2015-2016 were held and the output of the collaborative project will be a TECDOC series publication in 2018. Fortum and/or VTT will participate in the meetings in 2017.

WNA WG CORDEL is an industry sponsored group that promotes the achievement of a worldwide regulatory and industry environment where internationally accepted standardized reactor designs can be widely deployed without major design changes. Fortum participates in SMR ad hoc working group of CORDEL.

R&D activities within GIF are carried out at the project level and involve all sectors of the research community including universities, research institutes as well as industry [13]. Finland is a GIF participant via Euratom and many institutes and laboratories cooperate with GIF projects through exchange of information and results. VTT has been active in GIF SOWR Project Arrangement (PA) (e.g. Chair duties since 2013 in SOWR Materials & Chemistry Project Management Board) for last ten years and this collaboration will continue in 2017 in terms of Materials & Chemistry working group activities.

Finnish research organizations have participated in networking projects where the leading international research institutes coordinate their research on innovative reactor concepts. With this cooperation, combing the knowledges of e.g. structural materials for Gen4 or material improvements for Gen3/SMRs, manufacturing technologies, fuel technology and safety issues related to new fuel processing will spread the understanding of cross-cutting issues in the future nuclear reactor technology field in national level. In this task, the purpose is to aid participation in a) IAEA, b) WNA WG CORDEL, c) Nordic forums (e.g. workshops organized by Energiforsk or NKS) and d) GIF activities.
2.3 Work package 3 (WP3) Project management

The Nuclear Energy Act was amended in late 2003 to ensure funding for long-term nuclear safety and nuclear waste management research in Finland. The necessary finance is collected annually from the license holders to two special funds devoted to this purpose. The objective of the research funds is to ensure the high level of national safety research and to maintain the national competence in the long run [2]. For this purpose, GENXFIN is the forum where to discuss and coordinate the nationally important aspects in future reactor technologies.

The Finnish actors have achieved a significant role in performing and directing scientific research and technological development for Gen4 concepts in the global forums for the last ten years such as IAEA, EERA JPNM, ESNII and GIF. During the conceptual and design phases for the Gen4 demonstration plants Finnish partners have actively followed the development of different technological options. The recommendation of the YES strategy in 2014 [1] highlighted that the different research areas including future reactor technologies must be gathered into wide-ranging national programmes. Future nuclear energy technologies was seen important part of future research activities both nationally and internationally. It was also emphasised that the research programme on future reactor technologies should include a significant portion of different reactor technologies (SMR, Gen4, fusion) focusing on cross-cutting topics. Further to this, scope of the programme should be emphasised based on the reactor technologies that are of most important nationally. In general, this screening work on the most important research areas was done in 2015 - 2016 within the GEN4FIN and the follow-up project GENXFIN.

WP3 enhances the near-term actions in the research area of advanced reactor technology designs. The main goal of WP3 is the coordination of GENXFIN activities.

Table 5. Partners and person months allocated to WP3.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>0.5</td>
</tr>
<tr>
<td>Aalto (technical secretary, external services indicated in the resource plan, Annex 2-1)</td>
<td>0.1</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>0.5</strong></td>
</tr>
</tbody>
</table>
2.3.1 Task 1 (T3.1) Coordination of GENXFIN and information dissemination

The main objective of this task is to set up an effective coordination and management framework for the R&D work on advanced reactor technologies in Finland. In order to continue this activity, Finnish key players are gathered as consortium to ensure progress of the previous GEN4FIN network project towards its updated objectives and taking into account guidelines of main European and international forums and networks (e.g. through travel reports created in WP2). The GENXFIN project coordinator will work in close cooperation with the project managers in projects funded by EU and nationally. The purpose is also to enhance information dissemination between national partners. The idea is to share relevant documents through updated webpages e.g. travel reports as well as other relevant documents which facilitates communication between national partners and can give easy access to decision makers and safety authority to the latest achievements. Also updated GEN4FIN webpages will provide information on the upcoming events e.g. workshops and other events on future reactor technology topics.

This task consists of the following:

- Coordination of the project & R&D activities on future reactor technology topics
- Updating of the existing GEN4FIN webpages for information exchange
3. Deliverables 2017

Table 6. Deliverables for 2017.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Report on systems engineering approaches in licensing issues by Fortum and VTT</td>
<td>1.5</td>
<td>12/2017</td>
</tr>
<tr>
<td>D1.1.2</td>
<td>Report on severe accident management including radioactive release assessment in SMRs by VTT</td>
<td>1.5</td>
<td>12/2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Travel reports on IAEA, WNA CORDEL and GIF meetings by VTT and Fortum</td>
<td>1.8</td>
<td>12/2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td><strong>Milestone: update of existing GEN4FIN webpages to GENXFIN for information exchange</strong></td>
<td>0.5</td>
<td>5/2017</td>
</tr>
</tbody>
</table>
4. **Project organisation**

The project’s home organisation is VTT. VTT will act as intermediary between Finnish industry and international partners. The project coordinator is Mr Sami Penttilä / Mr Jarno Kolehmainen (Research Scientist) from VTT.

The project coordinator is responsible for the timely and effective execution of the work packages in accordance to the description of the project plan. The duration of the project is planned for 2016 – 2018. Project meetings will be held during the year when necessary, however at least one in the end of year. Possible conferences and workshops will be used for concerted actions and discussion.

Table 7. The following researchers are involved in different subtasks:

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Jarmo Ala-Heikkilä</td>
<td>Senior Research Associate</td>
<td>Aalto University</td>
<td>T3.1</td>
<td>0,1 (external services shown in Research plan, Annex 2-1)</td>
</tr>
<tr>
<td>Karoliina Ekström</td>
<td>Project Portfolio Manager</td>
<td>Fortum</td>
<td>T1.1</td>
<td>in-kind</td>
</tr>
<tr>
<td>Antti Rantakaulio</td>
<td>Design Engineer</td>
<td>Fortum</td>
<td>T1.1</td>
<td>in-kind</td>
</tr>
<tr>
<td>Ville Tulkki</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T2.1, T3.1</td>
<td>0,3</td>
</tr>
<tr>
<td>Riku Tuominen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.1</td>
<td>0,25</td>
</tr>
<tr>
<td>Jarno Kolehmainen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T1.1, T3.1</td>
<td>1,75</td>
</tr>
<tr>
<td>Sami Penttilä</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.1, T3.1</td>
<td>1,5</td>
</tr>
<tr>
<td>Mikko Ilvonen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1, T2.1</td>
<td>1,5</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>5,3</strong></td>
</tr>
</tbody>
</table>
5. Risk management

There are no foreseeable significant risks for the implementation of the project. Some uncertainties can be seen especially in the workload of certain key persons. This risk can be handled by managing workloads so that each participant has an alternate co-worker. In WP1, the main challenge is most probably the lack of enough detail technical data or poor data quality which can affect the end result of the analyses performed. Other risks in WP2 and WP3 are seen very small, since they focus mainly on networking. Political decision making may affect the research topics in focus. However, research focus is on the cross-cutting issues which are not dependent on one reactor concept. In addition, this project plan will be updated every year based on the Finnish industry needs and main European forum guidelines.

Table 8. Risk management plan which will be updated during the project on a periodic basis.

<table>
<thead>
<tr>
<th>Risk</th>
<th>Probability of occurrence</th>
<th>Potential impact on project success</th>
<th>Mitigation Plan</th>
</tr>
</thead>
</table>
| Lack of technical data or poor data quality | Medium                  | Medium                              | - Concentrate on cases where data is available  
- Only use records which have good quality basic data sets                        |
| Loss of key researcher         | Low                       | High                                | - Identify alternative resources in case of unexpected absence.  
- Ensure complete records of work are available at any point                       |
| Cooperation between Fortum and VTT | Low                       | Medium                              | - Arrange a project start up meeting in the beginning of the project.  
- Include periodic project meetings to follow progress                              |
6. References

## Expenses

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat&amp;supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Memb fee</th>
<th>Other</th>
<th>TOTAL VYR</th>
<th>Fennovoima</th>
<th>Fortum</th>
<th>TVO</th>
<th>Aalto</th>
<th>LUT</th>
<th>VTT</th>
<th>NKS</th>
<th>Other</th>
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</thead>
<tbody>
<tr>
<td>WP1 - Work package 1 Safety features of SMRs</td>
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<td>8</td>
<td>0</td>
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<tr>
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<td></td>
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<td>43</td>
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<td>8</td>
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</tr>
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<td>WP2 - Work package 2 International co-operation</td>
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<td>0</td>
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<td>0</td>
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<tr>
<td>T2.1 Task 1 International networking</td>
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<td>0</td>
<td>8</td>
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<td>22</td>
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</tr>
<tr>
<td>WP3 - Work package 3 Project management</td>
<td>0.5</td>
<td>6</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>2</td>
<td>0</td>
<td>1</td>
<td>9</td>
<td>2</td>
<td>1</td>
<td>0</td>
<td>2</td>
<td>0</td>
<td>5</td>
<td>0</td>
</tr>
<tr>
<td>T3.1 Task 1 Coordination of GENXFIN &amp; Information dissemination</td>
<td>0.5</td>
<td>6</td>
<td></td>
<td>2</td>
<td>0</td>
<td>0.8</td>
<td>9</td>
<td>2</td>
<td>1</td>
<td>2</td>
<td>1</td>
<td>0</td>
<td>2</td>
<td>0</td>
<td>5</td>
<td>0</td>
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<td><strong>TOTAL</strong></td>
<td>5.3</td>
<td>66</td>
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<td>17</td>
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<td>18</td>
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<td>0</td>
<td>48</td>
<td>0</td>
</tr>
</tbody>
</table>

### Comments:

- The "Other" costs include research infrastructure surcharge.
- WP1: Fortum in-kind (2 htkk). Fortum has own separate funding for this work in WP1 which is not shown in this resource plan.
- WP3: Technical secretary from Aalto. This is an external service and allocated in the resource plan (T3.1).
- Memb fee: Please explain the membership fees in international projects included in the cost budget (VYR funding 100% if approved).
- Describe possible in-kind work contribution here (organisation and person months, use of equipment etc.).
- Other explanatory comments.
MAPS

Management principles and safety culture in complex projects

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Jaakko Kujala, Kirsi Aaltonen
University of Oulu
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1. Research theme and motivation

Currently, various nuclear industry projects are carried out in Finland, including modernization projects in the operating plants, Olkiluoto 3 commissioning activities and the new build project Hanhikivi1. Such complex projects bring together a diverse set of actors with differing values, knowledge, cultures, practices, goals and business models. In general, complex projects\(^1\) are large-scale temporal undertakings, subject to high levels of uncertainty with major financial, environmental and social implications for project stakeholders and society. A vast number of large multinational projects, for example in the construction sector, have poor performance record in terms of economy and public support. Prior literature suggests that inability to manage the increasing complexity in projects may be a significant factor in deficient project performance (e.g. Morris & Hough, 1987; Flyvbjerg et al., 2003). The project governance challenge is to align and coordinate numerous interrelated project roles and activities, characterized by ambiguity of cause-effect relationships and difficulty to understand and control the behavior of the project network actors.

Projects in the nuclear industry, such as plant modernizations and new builds, are increasingly carried out by a multinational network of companies. Some of the project parties might have little experience in the Finnish regulatory requirements or nuclear industry practices in general. In this context, ensuring that the safety and quality requirements are adequately understood and fulfilled by every party is a challenging task. Many of the project participants work in other industries, or are committed also to other temporary endeavours, so it cannot be expected that they necessarily share the same values, knowledge and working methods that support the overall safety goal of the project.

The Nuclear Energy Agency (NEA, 2015) has recently highlighted that successful project management in the nuclear industry includes more than engineering excellence and financial skills: “soft issues”, such as “leadership, the development of cross-cultural understanding, team building, the creation of appropriate incentive structures and trust” are considered as important factors. Nuclear industry projects are also a subject to safety rules and regulations, which add on a dimension to effective management. In order to support safety in large and multinational nuclear industry projects, there is clearly a theoretical and practical need to advance the understanding on the links between management principles and safety culture.

There have been different challenges associated with schedule and quality in recent major projects in the nuclear industry (STUK 2011). Suboptimal project management and insufficient nuclear safety culture of the network, formed by the supplier and its subcontractors, have been suspected as contributing factors to these challenges (INPO 2010). Challenges in schedule and quality may reflect issues in knowledge, competence, information flow, roles and responsibilities and attitudes among the project participants. Delays may cause pressures to cut corners, \(^1\) As pointed out by Ruuska et al. (2011), a large complex project can be viewed as “a dynamic network of organizations that combines the resources, capabilities and knowledge of the participating actors to fulfil the needs of the owner”. The shared project goal, such as construction of a nuclear power plant, each actor in the project network is directed by its own goals, and these objectives might sometimes be in conflict. In the project literature different related terms are used to designate the most typical characteristic of such an entity, involving many organizational actors who should deliver a system of considerable size and complexity, such as complex project (Barlow, 2000), major project (Morris & Hough, 1987), giant project (Grün, 2004), megaproject (Flyvbjerg et al., 2003), large project (Miller & Lessard, 2001a, 2001b) have been suggested to describe projects.
create tensions between partners, deteriorate open communication climate, accelerate turnover of key persons and increase the risk of latent technical problems or non-conservative decision making. Therefore, the performance of the network of actors involved in various lifecycle stages of projects contributes to the defence-in-depth. Consequently, the Radiation and Nuclear Safety Authority in Finland (STUK) has recently issued new YVL guide with requirements on project management and safety culture of suppliers and subcontractors, e.g. YVL A.3 (STUK, 2014). The international nuclear institutions have also paid attention to project management and safety culture in networks (INPO 2010, The Royal Academy of Engineering 2011, IAEA 2012).

Still, safety research has so far paid little attention to project management because project delays and quality issues have been perceived mainly as economic problems and not safety concerns as such. However, safety cannot be separated from other performance issues when a systemic approach to safety is applied. In projects, actions of one organization can affect other stakeholders in a dynamic and somewhat unexpected way. As the IAEA (2014) indicated in a report on Human and Organizational Factors in Nuclear Safety in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, “a systemic approach to nuclear safety takes into account the dynamic interactions within and among all relevant factors of the system - individual factors (e.g. knowledge, thoughts, decisions, actions), technical factors (e.g. technology, tools, equipment) and organizational factors (management system, organizational structure, governance and human and financial resources)”.

Current safety culture and safety management models and practices are largely focused on a single organisation and it is far from clear how to apply them in dynamic temporary settings, such as projects. Some characteristics of complex project networks challenge the usability of the safety culture concept. For example, cultural approaches emphasize that it takes time and certain amount of continuity to create a culture, both of which are in short supply in projects with short time frames and often high personnel turnover. However, it is important to take into account that there are various project governance models. Project organising and management affects the daily activities and the overall safety culture. Therefore, to support safe and effective execution of complex nuclear projects, there is a need to incorporate network expertise and project governance knowledge into the safety culture and safety management field.

The practical challenge of managing the complex network activities is also a scientifically challenging one. As the scale and diversity of activities in a project network increases, a wide range of new phenomena emerge, which require different logics of control than a traditional hierarchical enterprise. Networks have been shown to exhibit characteristics of complex adaptive systems (CAS) with self-organization, non-linear interactions and polycentric control (Choi et al. 2001, Reiman et al. 2014, Oedewald & Gotcheva, 2015). These features challenge some of the basic assumptions that underlie traditional safety management approaches, such as pre-specification of the course of activities and expectation for clear communication and control structures. There is also a need to critically review the safety management practices, which underlie current project management models and approaches, including safety culture models and auditing methods.

A set of important questions remain unanswered: How responsibilities between partners should be defined? How to prepare for project risks so that safety is not compromised? How to evaluate in advance the safety effects of various ways of organising? How to deal with these effects during the project? How to handle the national culture differences? How to deal with the fact that as the complexity of the project increases, there will always be more

---

2 Ahola et al. (2014) carried out a literature analysis on the concept of project governance and its origins, and pointed out that there is a lack of a shared and universally accepted definition of project governance. The authors indicated that there are two distinct streams of project governance research: the concept of project governance has been viewed in the existing literature either as external or internal to a specific project, with the majority of sources seeing it as internal feature. For example, Ruuska et al. (2009) defined project governance as encompassing aspects such as project practices, the management principles of stakeholders, documentation procedures, communication practices and contractual arrangements.
surprises and unanticipated circumstances that require adaptive capacity? Related to the latter is the crucial question of how to balance between standardization and situational adaptation in megaprojects, which can never be standardized to a same degree as e.g. control room instructions in an operating power plant? This leads to the overall research question of the project: what are the safety management principles that should be applied in managing complex projects in the nuclear industry, and how these principles can be implemented in practice?

1.1 Background and state-of-the-art

In safety critical industries, such as nuclear power, oil and aviation industry, the operating companies are expected to establish a systematic way of managing safety of their activities and to develop a good safety culture. However, many activities in those domains are not carried out by the operating company itself but by a network of contractors and subcontractors. In nuclear power industry subcontractor companies are often used in maintenance activities, modernizations projects, as well as in design and construction of new nuclear plants. The activities conducted by subcontractor companies may involve both occupational risks to the personnel and overall system safety impacts. While the role of contractors has been analyzed in accident investigations, e.g. Challenger space shuttle explosion (Vaughan 1996, Rogers 1986), Deepwater Horizon oil rig accident (Bea 2011), Fukushima Daiichi nuclear power plant accident (The National Diet of Japan 2012), scientific research of subcontractors and safety is largely focused on occupational safety (e.g. Mayhew et al. 1997, Jaselskis et al. 2008) with few exceptions (e.g. Quinlan et al. 2013, Dahl 2013, Nesheim & Gressgård 2014, Albrechtsen & Hovden 2014). Challenges of preventing occupational injuries of subcontractor workers may be different from those of managing the activities in a subcontractor network so that the overall system safety is created and maintained.

Many of the practical concepts and models used for improving system safety embed an implicit assumption that the activity is carried out by one organization, or rather, that the organization which is carrying out the activity corresponds to a single company or legal entity. This is reflected in safety management system literature, where management system is usually seen as a company specific system, although there have been some discussions since 1990’s on safety management in systems (Hale 1997). In safety management studies, the analysis can focus on “activity or company” (Hale et al. 1997) or different levels of the system: group level, facility level or at corporate level (Wahlstöm & Rollenhagen 2013). Still, studies on multi-company safety management systems are scarce with some exceptions, such as Reniers & Pavlova (2013) study which concluded that in the chemical industry there is a need for establishing a “multi-plant safety culture”, which should take into account the multi-company context as opposed to a single-plant viewpoint.

The same is relevant also for the concept of safety culture. The concept originates from the organizational culture concept in 1980’s, which aims at explaining the success of companies (Peters & Waterman 1982, Schein 1984, Schein 1992). The company focus has been adopted in the safety culture tradition. The frequently appearing notions in safety culture literature, for example, “top management commitment”, “open communication”, “organizational learning” and “levels of organization” (e.g. Cooper 2000, Guldenmund 2000, Sorensen 2002) imply that safety culture models have been developed to grasp a culture of a coherent unit. Conceptual studies on safety culture seldom discuss explicitly the unit of analysis issues. Antonsen (2009) highlighted that safety culture studies seem to embody a harmonious view of the organization to be analyzed. What should safety management system or safety culture improvement program be like in an “organization”, which is actually a dynamic network of actors from different companies? How to utilize these concepts in network settings? These are practical challenges relevant for the nuclear new build and modernization projects.
In SAFIR2014 the project “Managing safety culture throughout the lifecycle of nuclear plants” (MANSCU 2011-2014) highlighted that safety management and safety culture approaches should take better into account the networked nature of work processes in design in order to improve the quality and management of safety in design activities in complex projects (Macchi et al. 2013, Macchi et al. 2014). The project results also identified the pressing need to merge safety culture research with project governance because this perspective frames how the nuclear project activities are organized and coordinated, and how culture is developing (Oedewald 2012, Oedewald & Gotcheva, 2015). MANSCU project emphasized that lifecycle phases of a nuclear power plant have different core tasks and typical challenges, which may require different safety management and safety culture approaches.

The knowledge gained in SAFIR2014 project SISIANS is relevant as well. It showed some national characteristics of safety regulation, such as trust norms. Trust has traditionally influenced the approach to suppliers and subcontractors. However, in the increasing internationalized context of nuclear power projects, where multiple foreign subcontractors and workers interact, demand of trust may be a source of possible misunderstandings and misuses since there can be a mismatch between demands of trust and preconditions of trust (Ylönen 2014). Hence, the effect of different cultural aspects deriving from national, organisational or suborganisational level on companies’ performance cannot be neglected.

During the past two decades safety science has increasingly utilized complex systems theory ideas to explain why activities evolve out of control and disasters happen. Safety critical organizations have been viewed as complex socio-technical systems (Reason 1990, 1997; Rasmussen 1997, Vicente 1999, Reiman & Oedewald 2007) and the activities are often characterized as involving uncertainties, multiple conflicting goals, non-linear action-outcome effects and dynamic self-adaptation, which makes them challenging to control. A central message of the complex system approaches for safety work has been that safety cannot be created by decomposing the system into components, which will then be improved one by one. Instead, we should strive for approaches, which allow us to understand the dynamics of the system behavior and develop system capabilities for coping with varying conditions (Dekker 2006, Hollnagel 2009, Nemeth et al. 2009, Leveson et al. 2006). Although the need to apply systems view on safety (including human, organizational, societal elements) has been recognized in the nuclear industry since the Three Mile Island accident, the safety approaches can still be characterised as fairly mechanistic and technically focused (cf. Oedewald 2014, Reiman & Rollenhagen 2013).

The safety field could learn from network studies and project governance disciplines, where similar development has taken place recently. During the past decade there has been a significant increase in the frequency, complexity and magnitude of large infrastructure projects, such as building of nuclear power plants, tunnels, rebuilding of city centers and new and extended communication networks both in the industrialized and developing countries. These projects are huge financial undertakings and policy-making scenes, bringing together a large number of both internal and external stakeholders such as investors, contractors, subcontractors, local interest groups, government organizations, local towns and communities, political decision makers, and environmental groups with differing values, knowledge, cultures, traditions, goals, and business models (Flyvbjerg et al., 2003). Complex projects are fundamentally about managing and harnessing the uncertainty inherent in the project-set-up and environment.

Project governance has become a subject of intense research attention over the past years (Ahola et al. 2014, Brady & Davies, 2010). Recent research on the governance of megaprojects has brought up conflicting results on the effects of different types of multi-party contractual arrangements in their potential to align the interests of various stakeholders, manage uncertainty and ensure the financial, societal and environmental performance (Ahola et al. 2014). In parallel, the fundamental properties and organizational arrangements that optimize the megaproject system’s capability to respond to the unforeseen and unexpected events that are common during their lifecycle have been debated (Floricel & Miller 2001). An emerging research stream is also starting to address complex projects as hybrid meta-organizations consisting of multiple stakeholders (Gulati et al. 2012) and examine the implications of different types of organizational structures and their evolution for megaproject performance (Lundrigan & Gil 2014).
System dynamics modelling is another relevant discipline, which could be beneficial for understanding and supporting the overall performance of a complex project network. System dynamics modelling is a methodology to study complex adaptive systems. In the system dynamics methodology, the focus is on uncovering the feedback mechanisms, time delays, and accumulations that cause certain dynamic behaviour over time in a system. VTT has experience in the system dynamics modelling of project dynamics in product development (Pesonen et al. 2008), project manufacturing (Fox et al. 2009) and engineering design (Ruutu et al. 2011).

VTT has experience of managing confidential client projects related to analysing the dynamics of large complex projects using system dynamics. The importance of simulation is emphasized in the system dynamics methodology as a way to gain a better understanding of a system than by verbal reasoning alone (Sterman 2000). In literature on system dynamics applications to project management it is argued that although many theoretical mechanisms of project dynamics have been identified (e.g. various ripple and knock-on effect of managerial policies), the use of system dynamics to inform real-life project management is still missing. Notably, system dynamics models could also be integrated with other project management tools (Lyneis & Ford 2007), and this will be taken into account in the MAPS project. While prior work on system dynamics has examined the interrelationships between work quality, project delays and cost overruns, the effects of these factors on safety have not yet been explored systematically.

1.2 Objectives and expected results

The ultimate goal of MAPS project is to enhance nuclear safety by supporting high quality execution of complex nuclear industry projects, including modernisations and new builds. Three objectives are formulated according to this ultimate goal, as follows (Fig. 1):

1. Identify the generic safety principles of managing complex projects in the nuclear industry.
2. Clarify the cultural phenomena in major projects and the influence of time, scale, governance models, and the diversity of the involved actors on safety culture, and thus on safety.
3. Develop practical tools and guidance to facilitate the management and safety culture of ongoing and planned major projects focusing on, for example facilitating communication, managing change, organising decision making and problem solving in unexpected situations, encouraging openness, and distributing knowledge and lessons learned.

Figure 1. The ultimate goal and objectives of the MAPS project (2015-2018).
MAPS is a multidisciplinary project which brings together expertise in safety culture, organization science, complexity approaches, project network governance, societal research on safety regimes and system dynamics modelling.

**Expected results:** MAPS project contributes to an advanced understanding of how cultural issues affect safety and project management, and generally to a broader understanding of the concept of safety culture and its applicability in temporary organizations. The overall expected result is advanced knowledge by developing a set of theoretical frameworks, guidance and practical tools for defining and assessing project management practices and safety culture enhancement and assurance in nuclear industry modernizations and new build projects.

1.3 Exploitation of the results

The results produced throughout the MAPS project are applicable for the regulator, the power companies, suppliers and subcontractors in the nuclear industry. There are multiple ongoing complex projects in the Finnish nuclear industry and usefulness of MAPS results is continuously discussed with the end users to ensure that topical challenges in ongoing projects are timely and adequately taken into account by researchers.

*Power companies* are continuously developing their management systems and aiming at improving their procurement, contractor selection, auditing, training, supervision and safety culture and human performance development practices, as well as their project management, design and authorisation processes. MAPS project provides important insights to support that work. The *power companies* can utilize the MAPS research results in their own project management and safety culture improvement efforts and self-assessments of safety culture. The knowledge it brings may be also beneficial for strategic decision making concerning contract arrangements and outsourcing/insourcing of activities.

The *regulator* can exploit the results in their oversight of safety culture in plant modification projects, as well as in the design, construction and commissioning stages of the new build projects, for example by taking into account best practices in other countries.

The results are of interests also to the *suppliers and subcontractors* involved in nuclear industry projects in Finland in terms of project management and application of various safety culture development and assurance methods in the project network.

MAPS project utilises domestic and international forums for disseminating the results, such as organizational factors and safety related conferences (e.g. ESREL, SAFETY2016 World conference, WOSNET Workingonsafety.net, Resilience Engineering conferences), FinNuclear seminars, Nordic workshops, IAEA meetings and conferences, cross-industry seminars (e.g. oil & gas industry, construction industry and nuclear industry), as well as complex project management forums and project business conferences (European Academy of Management (EURAM), International Research Network on Organizing by Projects (IRNOP) and international workshops.

Project results are disseminated as oral presentations and written publications such as intermediate report, final report, scientific journal articles, conference papers, presentations, workshops, as well as a Master’s thesis. Participative dissemination workshops and lectures will be organised to ensure that the knowledge gained will be timely utilised. Important means of sharing the lessons learned throughout the project is the collaboration between the researchers, power companies and the regulatory body representatives in specific work packages. A system dynamics model of certain topical aspects of a complex nuclear industry project will be developed and introduced. This model will be used as a base for a simulation game development for management, which will be introduced in 2018.
1.4 Appropriateness of the project to SAFIR2018 programme

MAPS is a research project that thematically fits well into the Research Area 1 (Plant overall safety and systems engineering), especially section 3.2.4.2 “Organisation, human and interest groups”, described in the SAFIR2018 Framework plan. The update to the Framework plan for 2017 call, approved at the SAFIR2018 Management Board on 18 August 2016 is also taken into account in the current plan. Accordingly, Table 1 indicates the SAFIR2018 framework plan topics and challenges, the update to the Framework plan for the 2017 call, and the contribution of MAPS project to the topics, addressed by a specific Work Package/Task.

Table 1. MAPS project relevance to SAFIR2018 framework plan and updated plan (italics added)

<table>
<thead>
<tr>
<th>SAFIR2018 framework plan topics and challenges</th>
<th>Contribution of MAPS project to the topics (Addressed by Work Package/Task)</th>
</tr>
</thead>
<tbody>
<tr>
<td>“Several organisations have an effect on the safety of nuclear power plants during all stages of the design, implementation and operation of the plants.”</td>
<td>MAPS develops new knowledge on how complex networked projects in the nuclear industry can be managed so that safety and other project goals are not compromised (WP1-WP5)</td>
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<tr>
<td>“SAFIR has been researching the concept of safety culture for many years. Tools inspecting the safety culture have been produced and the safety culture has been assessed with studies. The concepts and tools of safety management have been created for the needs of an individual organisation.”</td>
<td>MAPS deepens the understanding of safety culture beyond the needs of an individual organization towards projects, which involve many different parties. MAPS contributes to an advanced understanding of cultural issues and their influence on nuclear safety and project management, and generally to a broader understanding of the concept of safety culture and its applicability in temporary organizations (WP1-WP5)</td>
</tr>
<tr>
<td>“Next, research is needed on whether there are effective ways for strengthening and developing the culture in a network consisting of actors with various roles, such as licensees, plant and equipment suppliers, authorities etc. It is important to study how the management of nuclear safety and other risks meet in the nuclear business.”</td>
<td>In MAPS we conduct empirical case studies of safety culture improvement in Finland and Sweden. We develop practical approaches for safety culture development and assurance in complex project networks by examining the existing methods for safety culture improvement. The applicability of the results is not limited to a single licensee but to all stakeholders (T3.3) MAPS applies knowledge on project governance and safety culture to provide recommendations for enhancing project management practices and recognizing safety consequences and various types of risks related to e.g. changes in projects, contractual relations, etc. (WP1, WP3)</td>
</tr>
<tr>
<td>“In the coming decades, Finland will have nuclear power plants in several different stages of their life cycle. This will generate research needs for understanding the development of expertise and competence already during the design and commissioning of a new nuclear power plant and utilising the know-how of experts working in plants already in use.”</td>
<td>MAPS contributes to the long-term continuity of competence in Finnish nuclear industry by educating new experts for the nuclear field by involving young scientists (e.g. master’s degree and doctoral students), post-doctoral researchers, as well as working closely with a community of project business scholars and university students (WP1-WP5)</td>
</tr>
</tbody>
</table>
"Changes and construction projects are increasingly implemented so that broader wholes are divided into long work and delivery chains with participants possibly coming from many countries and organisations. This creates the need to research the actions of these operating networks as a prerequisite for safe operations."

In MAPS we develop a conceptual model of challenges in achieving a good safety culture in complex, dynamic and multinational project networks, in which multiple subcultures interact. Cultural complexity is described in terms of how the different cultures interact and how it affects the formation of a shared safety culture in projects. We also study national culture influences on safety culture evaluations with the DISC model (Design for Integrated Safety Culture) (T3.1, T3.2)

"Safety management affects the actions not only within the plants but also in the control of work and action chains and in arranging the collaboration between different parties. Factors essential to the management of the overall safety and significant differences between the management in the different stages of the life cycle should be identified [...] Research is needed for developing systemic models for assessing the interactions of different social, psychological, organisational, economic and technological factors in the field of nuclear power."

The collaboration between different project parties is in the focus of MAPS project. We develop theoretical advancement and practical tools for understanding project governance mechanisms relevant for different stages of the lifecycle (WP1). We also develop conceptual and practical knowledge on safety culture development and assurance methods in different lifecycle stages in projects (T3.3)

One of the novelty values in MAPS lie in the integration of theories and concepts from different disciplines (T5.1). The research addresses the interactions of different factors in the field of nuclear power by bridging expertise in safety culture, organization science, complexity approaches and systems thinking, project business, governance of complex projects, construction industry network management, societal research on safety regimes and system dynamics modelling (WP1-WP5)

<table>
<thead>
<tr>
<th>SAFIR2018 update to the Framework plan for 2017 call</th>
<th>Contribution of MAPS project to the topics (Addressed by Work Package/Task)</th>
</tr>
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<tbody>
<tr>
<td>&quot;A memorandum of an ad hoc planning group is available in the material for 2017 call The themes described in the memorandum have become topical after the Fukushima Dai-ichi accident and should be taken into account in preparing the proposals. Many of the themes are already dealt with in the on-going SAFIR2018 research projects.&quot;</td>
<td>A topical challenge, related to the Fukushima accident and reflected in MAPS project, is the need to understand the roles of various project actors for creating nuclear safety. This includes the operating organisation, design companies, expert organisations, headquarters, subcontractor companies and regulatory bodies as parts of the system. Also, the Fukushima accident pointed out that national culture may also play a role in the formation of societal structures and institutional arrangements which contribute to nuclear safety, such as the national regulatory regime and communication patterns between different actors. These issues are also tackled in MAPS (T3.2)</td>
</tr>
<tr>
<td>&quot;Case studies would be important in the organisation research.&quot;</td>
<td>In-depth case studies continue in the power companies in 2017-2018 to gain insights on organizational dynamics and safety culture in the Finnish nuclear industry projects (T1.4)</td>
</tr>
<tr>
<td>The operation of the emergency preparedness organisation and co-operation and interactions between different organisations (&quot;organisation of organisations&quot;, tendency of the whole community to adopt questionable</td>
<td>Co-operation and interactions between different organisations in projects from project governance perspective is one of the research areas in MAPS. Results could potentially have implications for aligning of different organizations and emergency preparedness (WP1-WP4).</td>
</tr>
</tbody>
</table>
Beliefs shall be continued and emphasised in the research.

**Potential new presently unknown threats for safety could also be hypothesised by any appropriate means.**

Uncertainty and complexity in projects are on the MAPS’ “research radar”. We seek to gain insights on potential new presently unknown threats for safety in projects as dynamic temporary settings by applying a systemic approach and paying attention to (emergent) complexity, which is associated with interactions among components of a system and between the system and its environment (WP1, WP4).

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**High scientific quality** is ensured by scientific publishing. Collaboration with national and international research groups and practitioners provides greater impact and validity of the research. The MAPS project maintains and develops expertise, which the authorities and the power companies can utilise should any concerns arise in modernisations or new build projects that relate to safety culture, safety management, quality management, subcontractor’s performance, project governance, regulatory approach to project oversight. Availability and applicability of the results will be ensured by regular interactive stakeholder workshops and practitioner-focused intermediate and final reports. Since significant share of the MAPS project is carried out in case studies with the licensees/power companies, the focus on real-life issues in nuclear industry projects also promotes collaboration between the researchers at technical support organizations and universities and nuclear industry practitioners.

**International collaboration and knowledge exchange** is supported by MAPS project team’s international connections and establishing of new partnerships throughout the project. We maintain good cooperation with the International Atomic Energy Agency (IAEA) and OECD NEA (Division of Human Aspects of Nuclear Safety), which ensures information exchange about topical issues and meetings/conferences and monitoring the current state in the nuclear industry.

Professor Jaakko Kujala from University of Oulu has been appointed a Visiting Professor at Stanford University, USA. The results of this collaboration in terms of simulation modelling for the use of different governance approaches in complex project networks can be utilized to better understand the links between governance approaches in safety critical projects/mega projects and improving nuclear safety. Through partners at University of Oulu, MAPS also maintains active international collaboration with eminent project management scholars, such as Prof. Derek Walker (Australia) and Prof. Tim Brady (UK).

MAPS is collaborating with the Nordic Nuclear Safety Research (NKS) by receiving research funding for the subproject SC AIM (Task T3.3 of MAPS), which has strengthen the collaboration with the nuclear industry in Sweden. The Swedish nuclear power companies act as information exchange partners by collecting data and sharing experience on safety culture assurance and improvement methods in complex projects.

The benchmarking of Norwegian oil industry and research collaboration with the University of Stavanger carried out at MAPS provided an important new perspective from other safety-critical industry to further refine the understanding of nuclear specific requirements and management of complex projects in the Finnish nuclear industry. We maintain contacts with the Norwegian oil & gas industry, the regulatory body and research organizations. There is also exchange of information with the EU/SAFERA project STARS (SocioTechnical Safety Assessment within Risk Regulation Regimes), which is managed by VTT. STARS project studies how regulators assess organisational safety in various safety critical domains in France, Finland and Norway.
1.5 Education of experts

MAPS project involves young researchers and scientists who have not been previously working in the nuclear field. The VTT’s safety culture and system dynamics experts work together with leading project business and project governance professors at Aalto University and University of Oulu. This allows integration of different perspectives, development of innovative ideas and interdisciplinary knowledge transfer. The empirical work in MAPS, such as case studies, allows a deeper understanding of real-life nuclear industry for the young scientists, facilitates networking and opens future collaboration opportunities.

In 2016 Matilda Starck successfully completed her Master’s thesis at Aalto University/Hanken School of Economics, Department of Management and Organization on the topic “Exploring key dimensions of project governance and their relation to nuclear safety: An explorative study of nuclear industry projects”.

Among the VTT researchers, Kaupo Viitanen is a young scientist aiming at pursuing a PhD degree. Sampsa Ruutu is a PhD student at Aalto University. Doctoral students from the partner organizations University of Oulu and Aalto University are involved in the project whenever possible.
2. Work plan

MAPS is a four-year project (2015-2018). A detailed work plan is developed annually based on the progress of the project and fine-tuned to meet the actual needs of the stakeholders. When planning the work for each year, the regulator and the power companies are approached with an inquiry to identify topical aspects to drive the research in MAPS that could provide valuable benefits to the stakeholders.

The overall research structure of the MAPS follows the logics that conceptual and empirical works interact in a meaningful way. Since one of the novelty value lies in the integration of theories and concepts from different disciplines, special attention has been paid at creating conditions and allocating time for the project group to learn from each other before implementing different models in case studies and developing practical recommendations to stakeholders.

Overall, the first two years of the MAPS project (2015-2016) included carrying out literature reviews, baseline interviews, conceptual analysis and preliminary modelling of cultural process and systems dynamics. These models and concepts are currently studied, tested and further developed through case studies in selected Finnish nuclear industry projects. In the second part of the project (2017-2018) we focus on development and validation of practical tools (e.g. system dynamics modelling tool for project management), workshops with the stakeholders, international cooperation, scientific publications and disseminating the results via different channels and forums (e.g. scientific conferences, IAEA meetings, international nuclear industry workshops). During 2018 the focus will be on crystallising the guidance and practical tools for defining and evaluating project management practices and safety culture enhancement methods for the nuclear industry modernisation and new build projects, as well as high-level scientific journal publication activities.

MAPS project has five work packages (WPs), each consisting of several work tasks, which have been identified as practically challenging, yet scientifically understudied topics. WP1 and WP5 are crucial for creating the shared view on management and safety culture principles of complex projects, as well as for dissemination, internal coordination, integration of insights and project management. In order for these two integrative work packages to achieve their goals, we need work packages that pay attention to the regulator’s role and benchmarking of Norwegian oil & gas industry (WP2) and the role of cultural complexity (WP3) in project management and safety culture improvement in complex nuclear industry project networks. WP4 focuses on system dynamics modelling, which is a practical approach to test and visualise the dynamics in a complex project. It supports MAPS project’s internal work, as well as produces a practical tool together with the stakeholders to best meet their needs.

The overall work package and task structure for the four years of MAPS (2015-2018) is depicted in Figure 2. In the work plan that follows we provide a short overview of research carried out in 2015-2016, followed by a general description of the work to be conducted in 2017-2018, and a detailed description of the tasks in 2017.
Figure 2. Overall work package and task structure of MAPS (2015-2018) and focus on 2017 tasks.

<table>
<thead>
<tr>
<th>MAPS work packages and tasks</th>
<th>2015</th>
<th>2016</th>
<th>2017</th>
<th>2018</th>
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<tbody>
<tr>
<td><strong>WP1 Characteristics of complex projects and safety</strong></td>
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<tr>
<td>T1.1 Typical project governance models</td>
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### 2.1 Work package 1 (WP1) Characteristics of complex projects from nuclear safety point of view

**General description for 2017-2018**

WP1 continues throughout the four-years duration of the MAPS project and has three main goals. First, research in this work package focuses on characterizing complex projects and analyzing features that are challenging from management of safety point of view. Second, research provides knowledge on existing project governance and contract arrangement approaches and discusses their usability in nuclear industry context. Specifically, collaborative contract arrangements are studied. Third, the baseline empirical work that has started in 2015 continues in by carrying out in-depth case studies at the Finnish power companies to analyze if the generic features and lessons learned are valid in the Finnish nuclear projects, and to illustrate how the characteristics of the complex project manifest in practice. Case studies continue for an extended period of time (2016-2018) and comparisons are made to derive a further understanding of processes and developments of collaboration and nuclear safety culture in projects.
In 2015-2016 the research in WP1 focused on carrying out a systematic literature review of typical project governance models for complex multi-firm projects, and discussed them from nuclear safety point of view. This work nurtured the process of building a shared understanding in the interdisciplinary MAPS research team for developing a framework that is utilized in the empirical case studies. In 2015 the research resulted in a conference paper presented at the 6th International Project Business Workshop, 19-20 November 2015, Trondheim, Norway. The working report “Characteristics of complex projects in the Finnish nuclear industry: Interview study of three cases”, based on nine interviews provided basic theoretical background of project complexity and brief descriptions of selected complex projects in the Finnish nuclear industry. The report identified the need to pay more attention to non-technical aspects of complexity (e.g. organizational, emergent, institutional) and their implications for safety, and this is taken into account in Task 1.4.

In 2016 the focus was 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 WPs 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 applied in a project influence the actors’ behaviour. This builds understanding of the practical relevance of project governance approaches with regard to enhancing safety. Specific results were a conference article presented at the EURAM conference and successfully completed Master’s thesis by Matilda Starck focusing on the governance of safety critical projects. The focus of the thesis was on exploring and examining both theoretically and empirically what are the key elements in project governance and how they affect nuclear safety. The thesis was evaluated as “Very good”.

A conference paper Kujala, J., Aaltonen, K., Gotcheva, N. and Pekuri, A. (2016) “Key dimensions of project network governance and implications for safety in nuclear industry projects” was presented at the European Academy of Management (EURAM), 1-4 June 2016, Paris, France. The objective of this conceptual study was to increase our understanding of how to build governance systems for nuclear industry projects to enhance safety. A revised version of the manuscript, focusing on governance in inter-organizational project networks and implications for safety, is currently under preparation for submission to the International Journal for Managing Projects in Business (IJPM).

In Task 1.2 we explored the challenges associated with different contractual arrangements and the applicability of project alliancing/collaborative contract arrangements to the nuclear industry context. We investigated the applicability of governance approaches related to project alliancing (a particular type of project delivery method based on strong inter-organizational integration) and its influence on safety in nuclear industry projects. Specific outcome was organizing the International cross-industry workshop for the Finnish nuclear industry on the topic “Relational contracting for improved network performance in complex projects”. The workshop was held on 24 May 2016 at Aalto University. Guest speakers were Prof. Derek Walker (RMIT University, Australia) and Prof. Tim Brady (Brighton University, UK). Discussions provided insights into experiences of different industries (e.g. infrastructure, construction, nuclear power) on project alliancing, as well as international experiences on relational project deliveries and possible implications for safety, and lessons learned on management of complex projects. Deliverables are the workshop presentation material and Executive Summary for practitioners.

In Task 1.4 work planning was done in close cooperation with the power companies and the regulator, based on their needs and specific challenges. The in-depth case studies in the power companies are underway: we carried out 4 interviews for the Fortum’s case study (3 with Fortum’s experts and one with a STUK’s expert) and 11 interviews for the Fennovoima’s case study. The analysis of the qualitative interviews is progressing, as well as additional data collection.
Partners and person months allocated to WP1 in 2017 are given in the table. WP1 leader is University of Oulu (Jaakko Kujala).

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<th>Partners in WP1</th>
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2.1.1 Task 1.1 (T1.1) Typical project governance models

This task was finalized in 2016.

2.1.2 Task 1.2 (T1.2) Contractual arrangements and effects on performance

This task was finalized in 2016.

2.1.3 Task 1.3 (T1.3) Finnish experiences: Interviews at TVO, Fennovoima, Fortum

This task was finalized in 2015.

2.1.4 Task 1.4 (T1.4) Case studies: Organisational dynamics and management of complex projects

The goals and research content of T1.4 in 2017 is to continue with conducting in-depth case studies on critical incidents/events in complex nuclear industry projects at Fennovoima, TVO and Fortum to explore how project complexity manifests as safety challenges and how project organisations are handling these issues. Critical incident approach is utilized in the case studies, and the scope of the case is determined by an unexpected event, which the project network handled successfully or unsuccessfully, and there is a need to gain insights on managing change and organising decision making and problem solving in unexpected situations. In a complex project environment “critical incidents” could be seen as events that involve different project partners, and that make a significant contribution, either positively or negatively to the general aim of the activity. From each organisation a specific project case is selected. The case studies analyze the actions taken (or not taken) thereof by multiple organizations and the effects on, for example schedule, organization, progress and contents of tasks and document, as well as potential safety consequences. This analysis brings a micro-level event/activity view on stakeholders, their actions, relationships, and the overarching joint culture and dynamics in complex projects.

Specific tasks in T1.4 in 2017:
- Conducting interviews in the power companies
- Summary of the interviews and data analysis
- Writing scientific publications
- Organizing workshops with the power companies to discuss findings of the case studies

The method: Different methods are utilized in the case studies (e.g. interviews, workshops).

The volume of the task in 2017 is 4,5 person months.
The task will be managed by University of Oulu (Jaakko Kujala) and carried out jointly by VTT, University of Oulu and Aalto University.

The outcomes in 2017 will be workshops and a scientific publication:

a) Workshops with the power companies to discuss findings from the case studies
b) A joint article (VTT, University of Oulu and Aalto University) to IRNOP conference (International Research Network on Organizing by Projects), Boston University, USA.
c) A manuscript of a scientific article will be submitted to a leading project management journal

The task will be finalized in 2018.

2.2 Work package 2 (WP2) Nuclear specific requirements for complex projects

General description for 2017-2018

WP2 will not be carried out in MAPS2017 due to insufficient funding.

A short description of research carried out in 2015-2016

In 2015-2016 the focus of the work was on benchmarking between the nuclear industry and the Norwegian oil & gas industry in terms of governance of safety and management of complex projects. In 2015 benchmarking concerned governance practices of Norwegian Petroleum Safety Authority and Finnish Radiation and Nuclear Safety Authority, whilst in 2016 the study focused on the oil & gas industry and the nuclear industry experiences and management of complex projects and related innovations. Both the regulatory and the industry perspectives are required to get better insight into the complexity and the recent developments in handling complex projects and safety.

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. We identified the challenges that the oil & gas industry in Norway has faced in managing complex chains of multiple stakeholders, as well as solutions that the oil & gas industry and the regulatory body (Petroleum Safety Authority) have created to tackle these issues. In addition, recent developments in the oil & gas industry in terms of handling of risk and safety are analysed, and comparisons were made with the nuclear industry in Finland.

In 2016 the case studies on complex projects in Norwegian oil & gas industry were based on documentary analysis and four interviews with representatives of the oil & gas industry. The results of this work indicated that Norwegian oil companies have mostly adopted hands-on strategy in managing their relationship with the contractors and sub-suppliers, due to failed projects based on the hands-off strategy (i.e. giving freedom to contractor to realize the project). Since the industry has taken responsibility and active role in managing its relationship with the contractors, the regulatory body in Norway plays minor role in that respect. This work resulted in an article manuscript.

A joint cross-industry seminar with representatives of the Norwegian oil & gas industry, Petroleum Safety Authority, the Finnish nuclear industry and the Radiation and Nuclear Safety Authority (STUK) was held in January 2017. In 2016 an oral presentation at the Society for Risk Analysis (SRA) and a full paper were presented at the European Safety and Reliability Conference (ESREL), which served as a basis for developing an article manuscript addressing lessons learned from benchmarking between the oil & gas and the nuclear industry.
2.2.1 Task 2.1 (T2.1) Regulator’s role in setting constraints for management of projects

This task was finalized in 2015.

2.2.2 Task 2.2 (T2.2) Benchmarking between the nuclear industry and the oil & gas industry

This task was finalized in 2016.

2.2.3 Task 2.3 (T2.3) International peer groups and agencies

T2.3 will not be carried out in MAPS2017 due to insufficient funding.

2.3 Work package 3 (WP3) Safety culture in complex networked organisations

General description for 2017-2018

A complex project can be considered a “melting pot” of different, partly overlapping cultures: occupational cultures such as engineering or management culture meets national cultures in a context of multiple organizational cultures, interacting under the umbrella of a network culture. This “cultural melting pot” plays a significant role as a determinant of the project network formation, relationships dynamics and outcomes, yet its characteristics and mechanisms have not been fully understood.

The work carried out in WP3 provides theoretical framing and description of the cultural complexity in terms of how the different cultures interact and to what extent it is possible to create a shared safety culture in such a context. In 2017-2018 research in Task 3.2 will focus on providing a deeper knowledge on how the variability in national cultures could affect the safety culture assessment done with the DISC safety culture model. In Task 3.3 in 2017 we continue our study on methods for enhancing and assuring safety culture in complex projects to provide practical recommendations for improving the applicability and impacts of methods for enhancing safety culture in the nuclear industry subcontractor networks.

A short description of research carried out in 2015-2016

Research in WP3 in the first two years 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. In Task 3.2 we carried out a literature review to clarify the concept of cultural complexity and existing frameworks, and to provide an overall picture of the recent quality and safety-related challenges, experienced in complex projects in the safety critical domain or other industries. The preliminary research findings on complexity and safety culture in project networks were summarized in an overview conference paper and presented at the IAEA International Conference on Human and Organizational Aspects of Assuring Nuclear Safety – Exploring 30 Years of Safety Culture.

In Task 3.3 we reviewed relevant literature to identify the practical methods to improve and assure safety culture. Six classes emerged based on what was the intended objective of the methods: training, facilitation of interaction and communication, development of organizational structures, improving commitment and participation, promoting the visibility of safety culture and direct behavioural modification. Most of the existing methods were focused on operational phase of the NPP lifecycle and were not specifically designed for use in project organizations. In the
main case study (Fennovoima) we focused on the use of Safety Culture Ambassadors Group as a safety culture improvement method.

Findings indicated that the Ambassadors Group can facilitate the development and assurance of safety culture by improving information flows in the organization, including enabling bidirectional communication, encouraging and facilitating participation of personnel and ensuring that safety is taken into consideration in various activities in the organization. In the information exchange partner organizations, a variety of methods were applied for safety culture improvement and assurance, including development of working processes, organizing training sessions, seminars and workshops (also with top management) and encouraging people to speak openly if they have safety concerns. A simple framework for evaluating the pros and cons of the safety culture improvement and assurance methods in projects was developed. A presentation and a group work titled “Improvement Methods in Safety Culture” were held at the HUSC Expert Group Meeting on Safety Culture on 28 September 2016 in Helsinki. An extended abstract for a conference paper entitled “How to Build ‘Adaptive Culture’ in the Nuclear Industry? A Practitioner’s dilemma” has been written, as well as an intermediate NKS report documenting the findings from the research work carried out in 2016.

Person months allocated to WP3 in 2017 are given in the table below. VTT’s work efforts include task T3.2 (2 pm) and task T3.3 (3 pm), while University of Oulu supports the work in T3.2 (0,5 pm). WP3 leader is VTT (Nadezhda Gotcheva).

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2.3.1 Task 3.1 (T3.1) Modelling cultural complexity and safety culture challenges in projects

The task was finalized in 2016.

2.3.2 Task 3.2 (T3.2) National culture and DISC safety culture model

The goals and research content of Task 3.2: since this task is planned for 2 years, during 2017 we will focus on literature review on national culture/institutional differences and understandings of safety of project network actors and in 2018 we will focus on the implications for project management and safety culture evaluations in project networks. In this task, we will analyse whether the basic premises included in the safety culture model DISC (Design for Integrated Safety Culture) (Oedewald et al., 2011; Reiman & Oedewald 2009), which is commonly used in Finland, are national culture bound. The DISC framework places most importance to certain cultural characteristics that are important for safety. Specifically, the research question is: How the variability in national cultures in nuclear industry projects could affect the safety culture evaluation conducted with the DISC model? Is the DISC safety culture model universally applicable or is it biased towards the Western/Nordic culture?

Recent studies provide support for the need for development of advanced understanding of national culture influences on safety culture evaluations and the project management process (NEA, 2016; Noort et al., 2015; Oedewald & Gotcheva, 2016; Shore & Cross, 2005). The Nuclear Energy Agency (NEA, 2016) indicated that “national culture is one of the elements that should be considered in fostering and enhancing an organisation’s safety culture. The effect of national culture to safety culture of an organisation is twofold. Firstly, the individuals working in an organisation always execute some features of their national culture (e.g. certain values or social norms) in their
work behaviour. Secondly, national culture is embedded in the societal structures around nuclear safety (e.g. legislation, education, roles of different stakeholders) which may affect the organisations’ activities to a great extent.”

While focusing on studying safety culture in a network of companies, Oedewald & Gotcheva (2016) highlighted the need to understand how the characteristics of a good safety culture manifest in practice and how to promote those characteristics across different national contexts. From a practical perspective, it means that the same safety measures may not be equally effective across groups from different national cultures. The Fukushima Daiichi accident pointed out that national culture may also play a role in the formation of societal structures and systems, which contribute to safety of nuclear power production, such as the regulatory regime and communication processes between different actors (The National Diet of Japan 2012, Kinsella 2013).

Complex project networks are nested in a particular institutional context, related to the culture of the host country, which also incorporates the national cultures of the key actors in the network. Institutional contexts are characterized by a set of principles that prescribe means for interpreting and acting, which are referred to as institutional logics (Thornton, 2004). Organizations experience institutional complexity when they have to handle pluralistic demands, such as goals, interests and practices, stemming from multiple institutional logics (Greenwood et al., 2011).

Specific tasks in T3.2 in 2017:

- Literature review on national culture/institutional differences and understandings of safety of project network actors (stakeholders), and related implications (how is this managed/responded to)
- Organizing a workshop with nuclear industry practitioners from different national backgrounds

Method: Literature review, workshop

Volume of the task in 2017 is 2.5 person months (2 pm VTT and 0.5 pm University of Oulu).

Task will be managed by VTT (Nadezhda Gotcheva) and carried out by VTT and University of Oulu (Kirsi Aaltonen)

Outcomes in 2017 will be as follows:

1) Workshop with experts from different national backgrounds to discuss links between institutional differences/national culture and safety culture in nuclear industry projects
2) A conference paper [title to be decided later].

The task will be finalized in 2018.

2.3.3 Task 3.3 (T3.3) Safety culture development methods in subcontractor networks

The goals and research content of Task 3.3 in 2017 is to identify and specify methods to improve, facilitate and assure safety culture in complex projects. This task continues from 2016 and is partially funded by the Nordic Nuclear Safety Research (NKS). The project is entitled “Safety culture assurance and improvement methods in complex projects” (SC AIM) and in 2017 will be carried out in collaboration with Tmi Teemu Reiman, Fennovoima, Fortum, the Royal Institute of Technology (Carl Rollenhagen) and partners from the Swedish nuclear industry (e.g. Vattenfall, Forsmark, OKG) as case organizations and/or information exchange organizations.

A basic premise of this subproject of MAPS is that there has been generally a lot of attention on diagnosing and evaluating safety culture but less attention has been devoted to its improvement. A second premise is that a complex project environment sets unique requirements to safety culture improvement due to e.g. multiple organizations interacting, diverse background of personnel, schedules and contract issues. The methods applied in operating power plants may not be fully applicable in temporary and dynamic inter-organizational project networks. Further, the long supply chains and the licensee’s responsibility to oversee the safety culture of the entire project network put more demands on safety culture assurance. Methods for improvement/facilitation of safety culture are, for example, the use of safety culture ambassadors, learning from experience, toolbox talks, pre- and post-job briefs,
cross-organizational working groups, and training. Methods for assuring safety culture can include auditing, self-assessment and independent assessment or questionnaires.

In 2017 Tmi Teemu Reiman (PhD, Adjunct Professor) participates in the project. Reiman’s participation in this project shall not serve as direct R&D for Fennovoima. Any conflicts of interest are avoided by defining Reiman’s responsibilities in this project in such a manner that Reiman’s contribution is beneficial to the regulator and the nuclear industry as a whole. Reiman’s role in the project is to serve as a scientific advisor for the theoretical part of the work and to act as modelling support for the synthesis of the project results into applicable safety culture tools and methods. Reiman’s contribution to the project aims to assure the top-level scientific excellence of this research project and to ensure the real-world applicability of the results created within this project. Reiman’s contribution will not involve direct development of the safety culture-related methods or structures utilized at Fennovoima. Reiman contributes to the tasks 3, 4 and 5 below. Reiman will not gain access to the data collected in the Fennovoima case study, nor participate in the analysis or dissemination of the Fennovoima case study.

Specific tasks of T3.3 in 2017:

1) Carrying out a follow-up study at Fennovoima on the implementation progress of Safety Culture Ambassadors Group
2) Conducting information exchange with additional organizations (incl. Forsmark, Fortum and OKG)
3) Organizing three researchers’ workshops on the following topics:
   a) How to build an adaptive safety culture in dynamic organizational environments;
   b) Safety culture improvement and assurance methods (partly in Stockholm);
   c) Safety culture methods and their underlying assumptions in the context of safety paradigms (in Paris together with J.-C. Le Coze and a group of invited young generation safety scientists)
4) Writing three scientific publications based on the findings from tasks 1-3
5) Development of new methods based on identified needs, potentially useful methods, and existing methods in workshops with the researchers and the case organizations
6) Pilot testing of the selected new methods in selected case organizations
7) Writing final NKS report and dissemination of results

The methods are case studies and expert workshops. Fennovoima FH1 NPP construction project acts as a full case study, Fortum (Finland), Forsmark-Vattenfall and OKG (Sweden) are information exchange partners.

The volume of the task in 2017 is 3 person months (additionally 0,75 pm work to be carried out by Tmi Teemu Reiman)

The task will be managed by VTT (Kaupo Viitanen) and carried out by VTT and Tmi Teemu Reiman.

The outcomes in 2017 will be as follows:

1) Researcher workshop (How to build an adaptive safety culture in dynamic organizational environments)
2) Researcher workshop (Safety culture improvement and assurance methods)
3) Researcher workshop (Safety culture methods and their underlying assumptions in the context of safety paradigms)
4) Conference paper: “Building an “adaptive safety culture” in a nuclear construction project – insights to safety practitioners”
5) Workshop paper and presentation: “Towards actionable safety science” [working title]
6) Scientific publication “Improving safety culture – what do we really know?” [working title]
   
   Note: We aim to revise the conference papers into journal articles or a book chapter.
7) Final NKS report on safety culture improvement and assurance methods in complex projects

The task will be finalized in 2017.
2.4 Work Package 4 (WP4) Applying system dynamic modelling in complex projects

General description for 2017-2018

System dynamics modelling is a methodology to study complex adaptive systems. The focus is on uncovering the feedback mechanisms, time delays, and accumulations that cause certain dynamic behaviour over time in a system. System dynamics models can be integrated with other project management tools (Lyneis & Ford 2007). Research and applications of system dynamics to project management can be generally grouped into the following categories: 1) post-mortem assessments for disputes and learning; 2) project estimating and risk assessment; 3) change management, risk management, and project control; 4) management training and education (Lyneis & Ford 2007).

In the MAPS project the aim is to utilise system dynamics modelling for illustrating and anticipating the complex effects of various ways of organising and managing the performance of the project network. The work in WP4 also focuses on developing a practical system dynamics tool for project management training. WP4 is a four year work package which starts by literature review in 2015 and moves on to developing a system dynamics model in a case study in 2016-2017. In 2018 the practical tool for project management training purposes will be developed.

Currently there are discussions on potential collaboration with Professor Terry Williams, University of Hull (UK), whose specific research interests in modelling project behaviour are internationally renowned and highly relevant for the MAPS project.

A short description of research carried out in 2015-2016

WP4 focused on a literature review of the use of system dynamics modelling of complex safety critical projects. The literature review provided an understanding of the existing uses of system dynamics to project management issues in general, as well as the applications of system dynamics for analysing and improving safety culture in order to focus the work in task 4.2. A working report “System dynamics modelling of complex safety critical projects – a literature review” was finalized in December 2015. The literature review indicated that key issues in the existing project management models, such as the number of undiscovered errors, have implications for safety, yet the current models are mainly discussed from financial perspective, focusing on cost overruns and schedule slippages. There are existing models that focus on safety culture related phenomena but these models mostly deal with operations, not development projects.

In summer 2016 a meeting at STUK was held to get feedback and input for the systems dynamics modelling. On 25 October 2016 a system dynamics modelling development workshop was held at Fennovoima to better understand the new build perspective when developing the SD model. The focus is on document handling; this is a topical issue in new builds since small delays and disruptions in document handling processes can cause complex projects to suffer massive cost and time overruns. The causal loop diagram of how governance and safety aspects related in a nuclear industry project has been prepared. A research project entitled “Delays in creating and handling design documents” has been finalized in December 2016 and is currently under revision for publication approval.

Person months allocated to WP4 in 2017 are given in the table below. WP4 leader is VTT (Joona Tuovinen)

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2.4.1 Task 4.2 (T4.2) Development and testing a simulation model for nuclear industry projects

The goal and research content of task 4.2 in 2017 is to continue the work on development of a system dynamics simulation model to support the document handling of a complex project and to better understand the implications on safety.

Specific tasks in T4.2 in 2017:
- Data collection to support modelling the document handling of a complex project and its implications on safety
- Organize workshops jointly with the other work packages, as well as with the regulator and the power companies. Data gathered in the other work packages will also be used in the model building.

Method: Work in this task is system dynamics model building and includes the standard phases of a modelling process (Sterman 2000): 1) problem identification, 2) generation of dynamic hypotheses, 3) model formulation, 4) model testing and validation, 5) preliminary policy formulation.

Volume of the task is 2.5 person months (2 pm VTT and 0.5 pm University of Oulu).

Task will be managed by VTT (Joona Tuovinen) and carried out by VTT and University of Oulu.

Outcome: The model and its development will be reported in a conference/journal article.

The task will be finalized in 2017.

2.4.2 Task 4.3 (T4.3) Development of management training tool

The task will be carried out in 2018.

2.5 Work package 5 (WP5) Integration and dissemination of the results

General description for 2017-2018
In MAPS the importance of continuous stakeholder engagement for dissemination and utilization of research results is recognized. The purpose of WP5 is to ensure good project management and build a cohesive project team to integrate the findings from different work packages and produce novel theoretical and practical outputs that are useful for the nuclear industry stakeholders and the scientific community.

In WP5 the focus is on summarizing lessons learned from activities carried out in the project and answering the overall questions of the project: what are the safety principles of managing major projects in the nuclear industry? How do various cultural phenomena manifest in complex projects, and how the concept of safety culture can be applied in such a diverse and dynamic setting? Finally, what kind of practical tools and methods are required for better management of complex projects? These principles will be developed iteratively and jointly by the MAPS project team during the course of the project and tested in case studies with the industry and regulators. The intermediate and final reports that will be written as part of WP5 will also be planned and developed jointly by the project team.

WP5 includes project management and reporting, as well as activities aiming at ensuring internal cooperation in MAPS project and events to disseminate results to the nuclear industry stakeholders.

In 2017 MAPS and CORE projects (SAFIR2018 RG1) are planning cooperation via a joint workshop to enhance the understanding of the human factors in management of complex projects, as well as to discuss possible applications of the HUMTOOL in project contexts. International cooperation: invited participation is a safety culture-related IAEA meeting to present MAPS results.

A short description of research carried out in 2015-2016
During 2015-2016 the MAPS project team focused on carrying out various activities for developing a shared understanding on relevant concepts, such as project governance, complex projects, safety culture, regulatory regimes, system dynamics modelling, etc. Several internal workshops were organized to integrate research efforts, discuss ideas in different work packages, create inclusive project climate and an integrated research team. The internal workshops have been very successful in providing multi-disciplinary approach on how to manage safety critical projects. MAPS dissemination seminar on 11 January 2016 presented findings and received industry feedback for the first year of the project by stimulating interaction and open discussion between the power companies, the regulator and the project partners. MAPS results were presented at the following international conferences and events:

- The 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

Person months allocated to WP5 in 2017 are given in the table. VTT (Nadezhda Gotcheva) is the WP5 leader.

<table>
<thead>
<tr>
<th>Partners in WP5</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>0.5</td>
</tr>
<tr>
<td>University of Oulu</td>
<td>1</td>
</tr>
<tr>
<td>Aalto University</td>
<td>2.5</td>
</tr>
</tbody>
</table>

2.5.1 Task 5.1 (T5.1) Dissemination and internal coordination

The goal of this task is to ensure that the knowledge of the different disciplines is effectively integrated and the high practical and scientific value achieved. Furthermore, this task aims at disseminating the findings of the project at industry-oriented seminars, student workshops, joint project workshops and scientific conferences. The task also supports international cooperation of MAPS researchers with the nuclear industry (e.g. IAEA meetings and other events) and the project governance academic community and project management practitioners.

Method: project meetings, researchers’ workshops, industry-oriented seminars, joint publications

Outcomes in 2017:

- Joint project workshop MAPS-CORE, task HUMTOOL: A Human Factors view on safety management principles for managing complex projects and exploring applicability of HUMTOOL in nuclear projects
- Participation at IAEA meeting to present MAPS project results and develop international connections

The volume of this task in 2017 is 2.5 person months.

Task will be managed by VTT (Nadezhda Gotcheva), carried out by VTT, University of Oulu and Aalto University. The task will be finalized in 2018.
# 3. Deliverables and milestones 2017

Deliverables and milestones for 2017 are listed in the table. Milestones are presented in **bold** text.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Finalized by October 2017</th>
<th>Finalized by the end of 2017</th>
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<tr>
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<td>1\textsuperscript{st} deliverable of T1.4: Workshops with the power companies to discuss the case studies findings</td>
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<tr>
<td>D1.4.2</td>
<td>2\textsuperscript{nd} deliverable of T1.4: A joint conference article to IRNOP (International Research Network on Organizing by Projects), Boston University, USA.</td>
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<tr>
<td>D1.4.3</td>
<td>3\textsuperscript{rd} deliverable of T1.4: Manuscript of a joint scientific article, based on a case study, to be written and submitted to a leading project management journal</td>
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<tr>
<td>D3.2.1</td>
<td>1\textsuperscript{st} deliverable of T3.2: Workshop with experts from different national backgrounds to discuss links between institutional differences/national culture and safety culture in nuclear industry projects</td>
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<tr>
<td>D3.2.2</td>
<td>2\textsuperscript{nd} deliverable of T3.2: Conference article based on literature review and empirical data analysis</td>
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<tr>
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<td>Milestone [NKS]: Researchers' workshop on how to build an adaptive safety culture in dynamic organizational environments</td>
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<tr>
<td>N/A</td>
<td>Milestone [NKS]: Researchers' workshop on safety culture improvement and assurance methods</td>
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<td>N/A</td>
<td>Milestone [NKS]: Researchers' workshop on safety culture methods and their underlying assumptions in the context of safety paradigms</td>
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<td>D3.3.1</td>
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<td>D3.3.2</td>
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<td></td>
<td>x</td>
</tr>
</tbody>
</table>

**Total pm** 15
4. Project organisation

- Project manager: Nadezhda Gotcheva, VTT
- Deputy project manager: Marja Ylönen, VTT
- Organisation responsible for the whole project: VTT Technical Research Centre of Finland Ltd
- Financial Assistant: Irene Pihlajamäki, VTT

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
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<tbody>
<tr>
<td>Nadezhda Gotcheva</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>Project manager, WP1 (T1.4), WP2 (T2.3), <strong>WP3 leader</strong> (Tasks 3.2, 3.3) <strong>WP5 leader</strong> (T5.1)</td>
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<tr>
<td>Marja Ylönen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>WP3 (T3.2)</td>
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<td>Joona Tuovinen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td><strong>WP4 leader</strong> (T4.2), WP5</td>
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<td>Sampsa Ruutu</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>WP (T4.2), WP5</td>
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<td>Kaupo Viitanen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td><strong>WP3 (Task 3.3 leader, NKS)</strong>, WP5</td>
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<td>Karlos Arto</td>
<td>Professor</td>
<td>Aalto University</td>
<td>Scientific support for the project in the field of project business</td>
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<td>N.N.</td>
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<td>Jaakko Kujala</td>
<td>Professor</td>
<td>University of Oulu</td>
<td><strong>WP1 leader</strong> (T1.4) WP4, WP5</td>
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<td>Kirsi Aaltonen</td>
<td>Assistant Professor</td>
<td>University of Oulu</td>
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5. Risk management

The most significant risks that are identified in advance are possible budget cuts, difficulties to integrate the know-how of the participating researchers and research organisations, key personnel changes in the project group and practical issues in the power companies that could cause delays to case studies.

Budget cuts have an effect on the depth and extensiveness of the research activities, and as such could affect the quality of research results. Budget cuts can increase the risk of producing superficial results that have little scientific value and use for industry practitioners. Reductions in the VYR funding mean bigger overall cuts since the other funding is proportional to VYR share. The work package structure of the project divides the project in such a manner that useful results can be produced even if another work package or task is removed. However, it could mean that some of the partners would not find the project relevant or motivating anymore and could withdraw from the study. The work tasks are structured largely in a non-sequential way which avoids cascading effects and allows the continuation of the project even if some tasks are skipped.

The project is very interdisciplinary and experts from various organisations participate in research activities. One of the main risks to collaboration is challenges of integration between various research organisations which may lead to silos and inability to fully realize the innovative potential of the multidisciplinary research team. This risk is anticipated and prepared for by means of allocating researchers from several organisations and disciplines to common research tasks and work packages, as well as preparation of joint deliverables (e.g. writing joint articles), which advances communication and creates space for innovation between the various parties. In addition, a separate work task in WP5 is formed to ensure the coordination between the parties and to avoid the formation of silos.

Key personnel changes may cause issues in research progress and coordination between the participants. Personnel change risks are mitigated by various means of distributing the responsibilities. Nominating a deputy project manager who works in close collaboration with the project manager limits the effects of project manager change. Each work package is led by different persons, and as much as possible by different organisations, which ensures that loss of a single person or organisation would not affect multiple work packages. Key personnel changes are also prepared for by ensuring that in each work package there are participants from different organisations and disciplines, and that the number of work tasks that involve only one person is minimal.

Issues related to case studies are identified. The main risk is that a case study process cannot be started due to lack of time of the case organisation or deficiencies in the coordination of the case study. This risk is managed by careful planning of the case studies in close cooperation with the case organisations. In addition, clear responsibilities of case study coordination will be defined. Each case study task will have a person in charge of the coordination in the project team and a representative at the case organisation.


Lindee P.H. (2014). Dilemmas in Risk Regulation. Experience from the offshore oil and gas industry.


# Management principles and safety culture in complex projects

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat&amp;supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Other</th>
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<th>Aalto</th>
<th>Jyväskylä</th>
<th>In-kind</th>
<th>VTT</th>
<th>Other</th>
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Note: All figures are in euros.
Comments: Estimated travel costs:
T1.4 and T3.2 Domestic trips (case studies) (500 €), IRNOP conference, USA (3000 €); T3.3 Participation in four workshops in Sweden and Paris (3000 €), WOSNET conference (2000 €); T4.2 Conference trip (3000 €)
T6.1 Participation in IAEA meeting, Vienna (1000 €), WOSNET conference (2000 €), domestic project meetings and workshops and other dissemination activities (500 €)

External services:
T3.3 is partially funded by NKS. The external services in 2017 for T3.3 (13 000 €) are justified by order of subcontracting services from Teemu Reiman for carrying out specific research tasks and scientific publications.
T6.1 Catering for RG1 meeting, organized by MAPS (300 €), organizing and catering for two practitioners' workshops (700 €)
# VTT

### Work packages and Tasks

**WP1 Characteristics of complex projects and safety**
- T1.1 Typical project governance models: 5 person months, 22 keuro
- T1.2 Contractual arrangements' effects on performance: 0 person months, 0 keuro
- T1.3 Finnish experiences: Interviews at TVO, FV, Fortum: 2.5 person months, 22.0 keuro
- T1.4 Case studies of organisational dynamics and mgmt: 23 person months, 19.5 keuro

**WP2 Nuclear specific requirements**
- T2.1 Regulator's role in setting constrains and requirements for projects: 0 person months, 0 keuro
- T2.2 Benchmarking Norwegian oil industry: 0 person months, 0 keuro
- T2.3 International peer groups and agencies: 0 person months, 0 keuro

**WP3 Safety Culture in complex network organisations**
- T3.1 Modelling the cultural complexity and sc in projects: 0 person months, 0 keuro
- T3.2 National culture and DISC model: 20 person months, 15.5 keuro
- T3.3 Safety culture development methods in networks: 70 person months, 35.0 keuro

**WP4 Applying system dynamics modelling**
- T4.1 Review of system dynamics applications to projects: 0 person months, 0 keuro
- T4.2 Development of testing of a system dynamic model: 22 person months, 18.0 keuro
- T4.3 Development of management training tool: 0 person months, 0 keuro

**WP5 Integration and dissemination of results**
- T5.1 Dissemination and internal coordination: 0 person months, 0 keuro

| Total | 10.0 | 120 | 0 | 10 | 14 | 0 | 144 | 69 | 0 | 0 | 0 | 35 | 0 | 40 | 0 | 144 |

**Comments:** Task 3.3 "Safety culture development methods in networks" is partially funded by NKS. VTT is the coordinating organization in the project, KTH and Tmi Teemu Reiman are the other participating research entities. There is also an in-kind contribution of the commitment of the case organizations (Fennovoima, Forsmark, Fortum and OKG).
<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
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<td>Volume</td>
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<td>Travel</td>
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| Work packages and Tasks                          | person month   | euro    | euro   | euro    | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro | euro 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<td>1</td>
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<td>T2.2 Benchmarking Norwegian oil industry</td>
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<td>T2.3 International peer groups and agencies</td>
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<td>T3.3 Safety culture development methods in networks</td>
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<td>WP4 Applying system dynamics modelling</td>
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<td>T4.1 Review of system dynamics applications to projects</td>
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<td>T4.2 Development of testing of a system dynamic model</td>
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<td>T4.3 Development of management training tool</td>
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Comments:
SAFIR2018 Project plan

PRAMEA
Probabilistic risk assessment method development and applications

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1. Research theme and motivation

Probabilistic risk assessment (PRA) is the field of quantifying risks in terms of probabilities, evaluating the contribution of different subsystems, processes etc. to total system risk, and assessing the uncertainty related to the analyses. Currently, as modernisation takes place (also) in the nuclear domain, the functional principles and practices both in the process and its control change. Modernisation raises the question of the existence of the needs to renew the practices and perhaps also the principles of PRA used in the nuclear power plant (NPP) in question. Furthermore, new builds in the nuclear domain call for tailored PRA.

The PRAMEA project will cover the important and topical issues in probabilistic risk/safety assessment for nuclear power plants. Its main goals are to:

- Improve and develop methods for risk-informed decision making to support strategic and operative plant management
- Improve and develop PRA methods in terms of uncertainties and critical areas
- Develop PRA knowledge and expertise in Finland
- Foster international co-operation and import the best practices of the field to Finland

1.1 Background and state-of-the-art

PRAMEA continues the work done in PRADA and FinPSA-transfer projects as well as HACAS related to HRA activities, and possibly also some activities of the SARANA project, in the SAFIR2014 research programme.

As a whole, probabilistic risk analysis has yielded certain maturity, and is a central part of the safety justification of nuclear power plants. However, major research issues still remain, for example:

- Despite 30 years of development and experience with human reliability analysis (HRA) for PRA, there are weaknesses in the methods, which unnecessarily decrease their credibility. The problem areas include e.g.:
  - impact of the digitalization of control rooms
  - rules to identify relevant human interactions and what can be screened out
  - how to define the context, which is the basis for the analysis (elements of context, level of details)
  - how to benefit more from the qualitative analysis
  - definitions and scales for performance shaping factors
  - how to account for dependencies
  - how to select between time-dependent and time-independent quantification models
- In the HACAS project, data on validation of new control room concepts concerning user interfaces and team work was found relevant and useful for HRA purposes. However, several issues were left open in this work, starting from the appropriateness of the methods, originally used for validation purposes and now for HRA purposes, to the relevance of all types of validation data for HRA purposes.

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1 Issues have recently been identified e.g. in the review of plant-specific studies (known weaknesses of present methods), in the international empirical HRA method benchmark study (Forester et al. 2013), conclusions from the Nordic-German EXAM-HRA project (Bladh et al. 2014), in the WGRISK/WGHOF evaluation of HRA methods (WGRISK & WGHOF, 2015), and in the application of the HRA requirements guidelines such as HRA Good Practices (NUREG-1792; Kolaczkowski et al. 2005) and ASME PRA requirements (ASME/ANS 2009).
Dynamic PRA approaches can be more suitable in some situations than the fault tree/event tree approach in the reliability analysis of dynamic systems. However, dynamic approaches are still mainly in trial stage and it may be difficult to apply them in full scale PRA.

In severe accidents research, there is still substantial uncertainty concerning the phenomena of accident progression, and consequences of radioactive releases. The Fukushima Daiichi nuclear accident has provided some important lessons that need to be taken into account in PRA.

The Fukushima Daiichi accident has also meant increased interest in multi-unit PRA. Major part of the nuclear power sites house more than one reactor unit and other nuclear facilities such as spent fuel pool storage. Currently, multi-unit risks have not typically been adequately accounted for in risk assessments, but there is internationally a lot discussion on approaches.

Software tools specifically designed to support level 2 PRA are not widely in use. Many level 2 PRAs rely on level 1 techniques which have limited capabilities for modelling dynamic events and dependencies of severe accident progression. A new Windows-based version of the FinPSA code has been developed in FinPSA-transfer project in the previous SAFIR programme and in this project.

Level 1 and 2 PRAs are currently used as practically separate entities even though level 1 PRA is an input to level 2 PRA. FinPSA tool contains an implementation of tight integration of level 1 and 2 PRAs, but to our knowledge, the tight integration has not yet been taken into use at any NPP. Without tight integration level 2 results are not traceable to level 1 and the importances of level 1 accident sequences cannot be estimated with regard to radioactive releases.

The risk analysis of emergency operations has received some attention, but is not yet a part of the PSA of nuclear instalments. Analysis of schedule risks is relatively well-understood, but work still needs to be done e.g. concerning the limited resources case. There is much research on cost risks in project management literature, but the optimal allocation of resources and cost-effective preparation for the operations has received relatively little attention. Performance risks (that the end product of the operation fulfills its specifications – e.g. that enough water is pumped to the reactor core in recovering from LOCA) have not received sufficient attention. Some other risks, such as occupational hazard risks to emergency workers (NPP personnel, fire brigades etc.) and their implications to e.g. schedule risks, haven’t received quantitative, probabilistic treatment in the nuclear safety context. Groundwork for the modelling and risk analysis of such operations was created in the PRADA project.

There is a clear correspondence between PRA levels and the levels of defence-in-depth (DiD) (Holmberg and Nirmark 2008). Therefore, PRA is a natural way of analysing the goodness of DiD, and finding weak links in the
DiD layers. From the DiD point of view, research in PRA is needed in many topics. Research on computational methods is needed because DiD is a system-level concept, and therefore models that take DiD meaningfully into account are generally large, causing a strain on computational resources. Research on single issues within PRA (e.g. human reliability, reliability of digital systems, phenomena of serious accident progression) is needed so that these can be incorporated in the analysis of DiD.

PRAMEA will address these issues, more specifically:

- Methods for HRA relative to modernised NPPs or new builds using digital user interfaces. Especially methods for the identification of HFEs typical of control room operations with digital HMIs and the identification of relevant PSFs regarding the digital user interfaces will be developed. The question of how to take advantage of qualitative HRA methods in design, validation and training of digital HMIs will also be considered.
- In severe accident modelling and analysis the challenges are also in today’s computation architectures. The research will take into account existing models, but support for better and more traceable modelling is developed, and Finnish knowledge is strengthened in this domain.
- In the consequence analysis of radioactive releases (level 3 PRA), the main focus will be on improved modelling, the interplay of different methods and computations in level 3 analysis, and on uncertainty propagation and assessment. Case studies will be conducted with the aim of shedding light to important level 3 issues.
- Approach to analyse the site level risk will be developed. This includes both development of suitable risk metrics for site risk analysis (or multi-unit PRA), and development of methods to assess risk for multi-unit scenarios.
- The quantitative risk assessment of preparation and emergency operations will be developed so that the major risks — schedule risk, and performance risk — can be taken into account in the PRA of nuclear facilities, and there will be methodical support for planning the carrying out of such operations in a more risk-informed and cost-effective way.
- Modelling of severe accident phenomena and safety systems in level 2 PRA.
- The computational efficiency and capacity of dynamic PRA tools.

1.2 Objectives and expected results

The main objectives are as follows:

- to enable a credible human reliability analysis for digitalized control rooms in an acceptable, reliable and unified manner, to obtain new information on other topical issues and to train new HRA experts.
- develop methods that enable the incorporation of the probability of an emergency operation success, and timing information, to the main PRA models of nuclear power plants, and the planning of more cost-effective emergency operations with optimal use of scarce resources.
- Improve and develop methods to support a more extensive and practical overall safety assessment, and improve algorithms for computational PRA.
- develop a deeper understanding of some important serious accident phenomena (e.g. steam and hydrogen explosions) and how their analysis connects to the general framework of level 2 PRA.
- develop methods of uncertainty handling on level 3 PRA, improved ways of modelling important level 3 issues (such as evacuation), and software to aid level 3 PRA analyses.
- develop methods for site risk analysis, including development of risk criteria for site risk analysis and methods to analyse multi-unit accident scenarios.
- develop methods and models for the analysis of schedule and end-product quality risks in emergency operations so that estimates can be calculated for PRA-relevant quantities such as success probability of operation and the timing of operation completion.
- analyse the risks associated with an organization performing activities to attain a goal; for example, the risk that an error in a plan will not be detected in reviews and other quality assurance activities, and will affect the implementation of the plan.
- study applications and computational issues of dynamic flowgraph methodology.
- train new PRA experts.
- develop cooperation with Finnish experts.
- foster international collaboration.
The expected results of the project are as follows:

- a framework for HRA that would take into account the digitalized HMIs in the control rooms in nuclear power plants. Specifically, the framework will cover methods for identifying relevant PSFs and the formation of HFEs in such a way that it supports not only the renewal of HRA but also can also be utilised in supporting operator work in the form of, for instance, training.
- research reports on the handling of some important topical issues in HRA
- methods (including guidelines and tool support when applicable) for specific areas of PRA (e.g. site risk analysis, time-dependent risk follow-up) to support more extensive, practical and efficient PRA.
- improved dynamic PRA and reliability analysis tools
- improved method support for level 2 analysis. The support covers integrated deterministic and probabilistic modelling and analysis possibilities
- method support for tight integration of PRA levels 1 and 2
- improved models and methods for uncertainty analysis on level 3, more accurate models of important level 3 topics (e.g. dose assessment, countermeasures such as evacuation and shielding)
- software for level 3 PSA analyses
- case studies and improved methods for the probabilistic analysis and optimization of emergency operations
- methodological groundwork for the analysis of defense-in-depth in organizations
- new PRA experts at least in HRA, PSA level 2, and PSA level 3. All tasks involve the improvement and deepening of the knowledge and skills of current experts.
- cooperation and contacts between Finnish experts in e.g. modelling and analysis of complex organizations and operations (VTT and Aalto University), and atmospheric dispersion modelling (VTT and Finnish Meteorological Institute)
- knowledge on the applications of dynamic flowgraph methodology
- participation in international cooperation: IAEA, OECD/NEA WGRISK, ETSON PSA.

1.3 Exploitation of the results

The results of the project will have many applications within the nuclear industry:

<table>
<thead>
<tr>
<th>Result</th>
<th>ways of utilization</th>
<th>utilization time frame</th>
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<tbody>
<tr>
<td>HRA framework for digitalized control rooms</td>
<td>may be utilized directly when expertise is needed in HRA renewal; this is relevant in modernisation projects as well as in new builds where, obviously, digital interfaces are to be used. The framework may be put to use e.g. when the deployment of OL3 becomes topical and whenever there is a need to renew current HRA</td>
<td>within few years</td>
</tr>
<tr>
<td>The qualitative HRA analysis</td>
<td>will provide valuable information for a number of applications such as operator training, control room validation and handling of the design extension scenarios in the safety assessment. Provides insight for the renewal or updating the current HRA, in the form of contextual factors affecting human performance such as performance shaping factors. Provides guidelines for use of HRA in non-PRA applications, including human factors engineering (HFE)</td>
<td>can be started during the project</td>
</tr>
<tr>
<td>The guidance for the assessment of dependencies in HRA</td>
<td>can be adapted to present HRA methods used in Swedish and Finnish PRAs. Guidelines and survey prepared in 2015.</td>
<td>can be started after the project</td>
</tr>
<tr>
<td>Validation based on the data from simulator tests</td>
<td>will give a concrete view on matters which should be taken into account in HRA in digitalisation projects</td>
<td>Within few years</td>
</tr>
<tr>
<td>Work related to existing level 2 models</td>
<td>can be utilized in a new modelling and analysis framework. Avoiding of remodelling hurries up validation of models in new environment.</td>
<td>can be started during the project</td>
</tr>
<tr>
<td>Level 1 method development</td>
<td>guidelines and/or tool support to be used in specific PRA areas and more powerful computational algorithms will</td>
<td>a couple of years</td>
</tr>
<tr>
<td>Knowledge on dynamic risk analysis methods</td>
<td>Dynamic risk analysis methods can be used to complement static methods (e.g. fault trees) in those areas of PRA where static modelling is too restricted.</td>
<td>a couple of years</td>
</tr>
<tr>
<td>tight integration between level 1 and 2</td>
<td>gives new information about impact of level 1 events on level 2</td>
<td>is ready to be utilized</td>
</tr>
<tr>
<td>Case studies in level 2 PRA</td>
<td>shed light on some important phenomena in accident progression, and will be useful in the assessment of containment integrity, timing of events, and source terms. The end users are the level 2 PRA experts of the NPP companies and regulatory bodies.</td>
<td>after each case study has been completed</td>
</tr>
<tr>
<td>Methods for taking external events into account in level 3 PRA</td>
<td>Taking external events into account in level 3 PRA, either as initiating events or as events that affect accident consequences, improves the accuracy and plausibility of level 3 PRA analyses.</td>
<td>a couple of years</td>
</tr>
<tr>
<td>Methods developed for the analysis of preparation and emergency operations</td>
<td>may be used in assessing the success probability of operations in PRA models of NPP’s. They also aid the planning of prevention and emergency operations.</td>
<td>as soon as they have been developed.</td>
</tr>
<tr>
<td>Approach to site level PRA</td>
<td>Guidance and common Nordic understanding for the assessment of multi-unit scenarios and presentation of results from site-level risk point of view</td>
<td>2018</td>
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1.4 Appropriateness of the project to SAFIR2018 programme

The project would, in its part, fulfil many of the research needs in PRA presented in the SAFIR 2018 framework plan.

Ensuring the safety of nuclear power plants can generally be presented with the principles of defence in depth: the preventive level, the protective level and the mitigating level. Nuclear safety will be secured only if these principles are adhered to in the technical design and in the actions of organisations and humans (SAFIR2018 Framework plan p. 29; SAFIR2018 runkosuunnitelma p.36). With respect to the actions of the operating personnel, it is important to study what kinds of changes new technology, in the form of, e.g., digital procedures causes in work practices and possibilities for conducting work in a safe and efficient way (SAFIR2018 Framework plan p.38; SAFIR2018 runkosuunnitelma p.48). HRA work package (WP2) will take care of HRA development, harmonizing and keeping HRA practices updated. PRA method development will provide novel and improved approaches for the assessment of defence-in-depth from different view points including e.g. structural defence and automation in addition to serious accident management and large release (SAFIR2018 framework programme p. 48). The development will also help to maintain computational PRA abilities and skills (SAFIR2018 framework programme p.49).

Multi-unit PRA is a field of topical importance due to the Fukushima Daiichi accident where the tsunami cut off the electricity from all units (except batteries of unit 3).

The case studies on level 2 may contribute to the improved understanding of some issues that have risen in connection with the Fukushima Daiichi nuclear accident; in units 1-3 of Fukushima Daiichi, melted fuel fell to and subsequently penetrated the bottom of the reactor pressure vessels, resulting in molten-fuel-concrete interactions beneath the pressure vessels that further increased the pressure within the containment (UNSCEAR 2013). Improved understanding on melt-coolant interaction and its effects on pressure load in the containment would have direct implications to the assessment of leak tightness of the containment, which the framework plan (p. 36) names as central in serious accidents.

Method development for level 2 PRA responds to the need identified in the framework plan (p. 45) that “PRA level 2 computation still requires the transfer of competence, development and research”. Integration of levels 2 and 3 is an answer to the need “interfaces must be included for possibly level 3 PRA” (p. 45).

The software development on level 3 PRA will, in the long term, contribute to the goal of assessing the implementation of the goals set in the Council of State Decree VNa 717/2013 “through independent calculation methods” (p. 45).
The research on defence-in-depth in organizations is groundwork and a partial answer to the research need identified in (SAFIR2018 framework plan, p. 32) that “to form a framework for the assessment of overall safety in which the traditional concept of defence in depth is clarified by integrating safety maintenance structures, ... and the actions of organisations and humans in a single model”. The mathematical and systems approach to be adapted surely adds value to research on these issues. Furthermore, the research proposed in WP8 generally contributes to the understanding of “the operation of the organisations and the interactions between them as factors contributing to the overall safety” (p. 30).

1.5 Education of experts

New Finnish experts will be trained in at least human reliability analysis, reliability analysis of systems containing digital subsystems, level 1-3 PRA, and the risk analysis of organizations and operations. It is also possible that a new expert will be trained in the reliability of systems containing digital subsystems.

Expertise will be enhanced and maintained in the fields of computational methods and level 2 PRA, and possibly also dynamic reliability analysis.

At least one MSc thesis will be done during the project.

The research will support dissertation work of several PhD students. At least one doctoral dissertation is expected during the project.

Additionally, expertise on HRA will be raised in VTT by benefitting the SAFIR2018 context in gathering knowledge and experiences so that more researchers are educated to understand HRA to the point they are knowledgeable enough to guide the representatives of NPPs in HRA related matters.
2. Work plan

The total volume of the project in 2016 will be 26.5 person months (VTT), 2.6 person months (Risk Pilot), and 11.5 person months (Aalto University), for a total of 40.6 person months.

2.1 Coordination and project management (WP1)

This workpackage covers project management, coordination and cooperation efforts within the project. Within Finland, main cooperation will occur between PRAMEA and related projects within the SAFIR2018 research programme (e.g. FIRED, SAUNA, EXWE).

Communication between the project staff and end users is facilitated by arranging meetings (FinPSA end user meetings, HRA workshops etc.).

International cooperation is conducted on the European level within IAEA, OECD/NEA WGRISK and ETSON PSA, and Aalto University has cooperation with Politecnico di Milano (Italy). In the Nordic context, cooperation will occur within NKS and NPSAG, where collaboration takes place with Nordic partners (IFE Halden in HRA, Lloyd’s Register Consulting within Multi-unit PRA, Kungliga Tekniska Högskolan and Lloyd’s Register within level 2 PSA, Lloyd’s Register, Vattenfall and ÅF Consult within level 3 PSA). A project partner, Risk Pilot, is a Nordic company, and therefore Nordic cooperation is a natural part of its business.

Project managerial duties include coordination, progress monitoring, reporting, and planning of following years.

<table>
<thead>
<tr>
<th>Partners in WP1 (project management)</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT Risk management, senior scientist</td>
<td>1 pm</td>
</tr>
<tr>
<td><strong>total</strong></td>
<td><strong>1 pm</strong></td>
</tr>
</tbody>
</table>

2.2 Human reliability analysis (HRA) (WP2)

**Background**

Human performance has large impact on the reliability and safety level of complex technical systems. For this reason, human reliability analysis (HRA) is an important part of PRA. Adequate analysis of human interactions is one of the elements to understand accident sequences and their relative importance to overall risk. HRA is a powerful tool both for PRA applications and non-PRA applications. The latter refers e.g. use of HRA as part of human factors engineering (HFE) as required in NUREG-0711 (U.S.NRC 2012).

Human reliability is context-dependent. HRA methods should be able to assess various contexts to evaluate the impact of changes in the context, such as control room ergonomics. The modernisation of nuclear power plants, often in the form of digitalisation and including the one of control rooms, as well as new builds, call for the renewal of HRA methods to account for changed human-system-interface (HSI).

In the nuclear sector, the US NRC Human Factors Engineering Program Review Model NUREG-0711 (U.S.NRC 2012) is the main reference document followed in plant modernisation and new-built projects. NUREG-0711 recognises the “integral” role of Human Reliability Analysis (HRA) in Probabilistic Safety Assessment (PSA) (U.S.NRC 2012, p. 43). The standard also clearly states that HRA should form part of the HFE process: “A HRA evaluates the potential for, and mechanisms of human error that might affect plant safety. Thus, it is an essential feature in assuring the HFE program goal of generating a design to minimise personnel errors, support their detection, and ensure recovery capability” (U.S.NRC 2012, p. 43). In practice, however, HRA is often not linked to the HFE or design process at all, and there is little guidance available to support the analyst in performing HRA in a non-PSA context.
WP Work Plan

In WP2, methodological improvements for HRA concerning advanced control rooms and new HRA applications will be defined during the years 2015-2018. In addition, there will be a line of yearly changing current topical issues. Some issues rely on methodological research and reflection, without being dependent on any specific nuclear power plant, whereas others are performed as case studies in which acquired knowledge is used and new knowledge will be formed.

Additionally, part of the work will be in the forms of cooperation within IAEA and training new experts for the area of HRA.

2015 - 2016

In 2015, WP2 was composed of three tasks and in 2016 of two tasks. Their main results are summarized below under three themes:

1. HRA for advanced control rooms
   - 2015, T2.1:
     i. The state of the art of HRA for advanced control rooms was studied by VTT. It was found that some methodological progress during recent years has been performed mainly by Korea and China, including proposals for new performance shaping factors and typical error types when using soft controls. However, the empirical validity of the proposals remains to be shown.
     ii. A workshop was held with participants from all Finnish nuclear power plants and STUK. The participants presented their current state and wishes concerning the HRA of digital control rooms as well as their control room development plans.
     iii. The framework scope and requirements for the coming years was refined.
   - 2016, T2.3:
     i. Presentation and proceedings paper at 39th Enlarged Halden Programme Group Meeting, 9 May, 2016, Fornebu, Norway, title: "Recent Development and Future Prospects in Human Reliability Analysis of Advanced Control Rooms in Nuclear Power Plants".
     ii. Evaluation of Performance Shaping Factors proposed in literature
     iii. Related activity: Preparation of EU H2020 Euratom proposal “HRA in RDM”. VTT is WP leader of WP5 Topical analysis on advanced human-system interfaces (HSI)

2. Qualitative HRA and interaction with Human Factors Engineering (HFE)
   - Benefits from qualitative analysis (2015: T2.2):
     i. The methods used in Sub-System validation were scrutinised as such and their suitability as inputs to HRA was contemplated. Especially plant performance measures, from the viewpoint of control-room operators, and measures for personnel (control-room operator) task performance were identified as useful also for HRA purposes. Furthermore, the very same methods were found adequate for other HFE purposes such as training, EOP (Emergency Operating Procedure) evaluation/development and operator ConOps (Concept of Operations) evaluation/development.
     i. A survey was carried out among Nordic stakeholders (utilities, regulators and consultants to find out experiences and ideas regarding use of HRA in non-PSA applications (Holmberg & Linasuo 2017). Most use of HRA is still within PSA, and the applications are typical, such as development of instructions, operator training, control room design (validation) and occurred events analysis. Several difficulties in using HRA in a non-PSA context, e.g., limited resources, limited project budgets, cross-organizational activity, lack of guidance

3. Dependencies in HRA (2015: T2.3)
   - Existing methods were examined and supplementary guidance was developed for an improved assessment of dependencies in HRA. The need for additional method development was also assessed. The task was performed in collaboration with a Nordic PSA Group
(NPSAG) project, which was carried out by Lloyd’s Register Consulting. The NPSAG project performed a literature study and case studies in Sweden. PRAMEA performed two case studies in Finland (Porthin 2015, Holmberg & Jacobsson 2016).

There has also been regular communication with IFE Halden regarding common interests and future cooperation. As a result, joint NKS applications have been submitted both to the 2016 call (rejected) and 2017 (under review, see 2017 task T2.1). In addition, discussions on regarding usage of HRA data from simulator training is ongoing.

2017

The PRAMEA WP2 work to be performed during 2017 is done within two tasks: T2.3 Requirements for HRA for advanced control rooms and T2.4 IAEA Safety Report on Human Reliability Assessment for Nuclear Installations. These tasks are described in a more detailed manner in the next sections, focusing on the study to be performed during 2017.

Partners and person months allocated to WP2 year 2017 are described in the tables below.

<table>
<thead>
<tr>
<th>Partners in WP2, total</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT Risk management, senior scientist</td>
<td>2.1</td>
</tr>
<tr>
<td>VTT Risk management, research scientist</td>
<td>1.5</td>
</tr>
<tr>
<td>VTT Human factors in complex systems</td>
<td>0.7</td>
</tr>
<tr>
<td>Risk Pilot, PRA/HRA expert</td>
<td>1</td>
</tr>
<tr>
<td><strong>total</strong></td>
<td><strong>5.3</strong></td>
</tr>
</tbody>
</table>

2.2.1 Human Factors and Human Reliability Integration at Nordic Plants (HR&R) (T2.1)

No activity in 2017 due to negative funding decision from NKS.

This task was aimed to continue the work in PRAMEA in 2015 task T2.2 (Benefits from qualitative analysis) and 2016 task T2.1 (HRA outside the PSA: State-of-the-practice survey). The task was planned as a cooperation between VTT, Risk Pilot and IFE, with joint funding from SAFIR, NKS and IFE own funding.

2.2.2 Requirements for HRA for advanced control rooms (T2.3)

This task continues the work in PRAMEA in 2015 task T2.1 (Framework for the HRA of digitalized control rooms) and 2016 task T2.3 (Requirements for HRA for advanced control rooms).

Control room Human-System-Interface (HSI) design is one of the most important elements in the human factors engineering program for new plants and plant modification projects. Digital HSIs are becoming common through modernisations and new-builds. This impacts the work of the operators in several ways: the working environment changes, new tasks emerge and the group dynamics and communication are modified.

Most of the existing HRA methods do not address the new aspects introduced by digital HSI, as confirmed e.g. by the OECD WGRISK/WHGOF Task Group on Establishing Desirable Attributes of HRA Techniques for Nuclear Safety in 2015 WGRISK & WGHOF (2015). The state-of-the-art of HRA for advanced control rooms was studied by VTT in PRAMEA in 2015. It was found that some methodological progress during recent years has been performed mainly by Korea and China, including proposals for new performance shaping factors and typical error types when using soft controls. The empirical validity of the proposals remains to be shown.

The main objectives of this task are to improve (a) the task analysis, and (b) the treatment of PSFs for computerised HSIs in modernised and new control rooms.
The main steps in this task are:

1. Literature review on key characteristics and trends for digital HSI, including computerized procedures, in modernized and new control rooms.
2. Identification of the specific tasks and errors that describe the work process in advanced control rooms for improving the HRA qualitative analysis / task analysis.
3. Establishing Performance Shaping Factors (PSF) unique for digital HSI, using e.g. background from literature and own previous research.

During 2017 the analysis of PSFs will be continued and the task analysis initiated. Special aspects of hybrid control rooms with digital and analogue technology will be accounted for.

VTT will submit a paper to and participate in the PSAM Topical Conference 2017 on Human Reliability, Quantitative Human Factors, and Risk Management, to be held in 7 – 9 June, 2017 in Munich, Germany. The abstract to the conference has been approved. VTT has also submitted an abstract to the Special issue on Foundations and novel domains for Human Reliability Analysis of the journal Reliability Engineering and System Safety.

The task will be performed by VTT. In 2017, the volume of the task is 4.3 person months. Task leader: Markus Porthin (VTT Risk management, senior scientist).

<table>
<thead>
<tr>
<th>Partners in T2.3 (Requirements for HRA for advanced control rooms)</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT Risk management, senior scientist</td>
<td>2.1</td>
</tr>
<tr>
<td>VTT Risk management, research scientist</td>
<td>1.5</td>
</tr>
<tr>
<td>VTT Human factors in complex systems, senior scientist</td>
<td>0.7</td>
</tr>
<tr>
<td>total</td>
<td>4.3</td>
</tr>
</tbody>
</table>

2.2.3 IAEA Safety Report on Human Reliability Assessment for Nuclear Installations (T2.4)

IAEA has decided to initiate an effort to prepare a Safety Report on Human Reliability Assessment (HRA) for Nuclear Installations. The initial consultants’ meeting will take place in November 2016, i.e., after the submission of 2017 SAFIR project application. Dr. Jan-Erik Holmberg has been invited from Finland to participate in this work. The tentative working plan for 2017–18 is the following:

- November 2016 – Q1 or Q2 of 2017: preparation of a draft of corresponding chapters of the Safety Report on HRA
- Q1 or Q2 of 2017: 2nd consultancy meeting to discuss the available draft of the Safety Report and polish it to be ready for review
- 13-17 November 2017: a large technical meeting here in Vienna with participation of HRA/PSA experts from different member states. The produced draft Safety Report will be distributed for comments of meeting participants before the meeting
- Q1 or Q2 of 2018: after technical meeting, it is planned to have 3rd consultancy meeting to finalize the document based on the comments received during the TM
- Q4 of 2018 – target date for publishing the Safety Report on Human Reliability Assessment for Nuclear Installations

In addition, IAEA has the ambition to prepare a kind of scientific paper summarizing the common vision and describing the Safety Report to be presented during a conferences such as PSAM, ANS PSA or ESREL.

The task will be performed by Risk Pilot Ab. In 2017, the volume of the task is 1 person months.

<table>
<thead>
<tr>
<th>Partners in T2.4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>RiskPilot, PRA/HRA expert</td>
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<tr>
<td>total</td>
<td>1</td>
</tr>
</tbody>
</table>
2.3 PRA level 1 method development (WP5)

This work package focuses of PRA level 1 method development. The goal of this task is to develop methods to facilitate the assessment of overall risk. During four years of SAFIR2018 this work package contains topics on quite specific issues (e.g. multi-unit PRA, dynamic reliability analysis and development of computation algorithms) that are not part of the other work packages.

<table>
<thead>
<tr>
<th>Partners in WP5</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT research scientist</td>
<td>2.1</td>
</tr>
<tr>
<td>Risk Pilot, PRA expert</td>
<td>4</td>
</tr>
</tbody>
</table>

2.3.1 Site level PRA (T5.1)

The goal of this task is to develop and compile the state-of-art methodology on how to perform site level risk analysis. This includes development of risk criteria and safety goals for a multi-unit risk assessment compared to today’s single unit based risk metrics, and development of analysis and modelling methods to handle the dependencies between the different units in a proper manner.

In 2015, the state-of-the-art regarding multi-unit PRA (MUPRA) modelling was reviewed (Björkman & Tyrväinen 2015). In 2016, a methodology for preliminary probabilistic multi-unit risk assessment was outlined (Tyrväinen et al. 2017). The methodology aims to estimate multi-unit core damage frequencies (or large early release frequencies) related to different multi-unit dependencies. The idea is to utilise existing PRA models of individual units as much as possible. The methodology contains phases for identification, analysis, modelling and quantification of multi-unit dependencies, but several parts were outlined only in conceptual level.

From 2017 forward, this task will be a part of a joint Nordic project called SITRON. The following plan concerns the whole SITRON project. During 2017–2018, the first objective with the task is to search for practical approaches for Nordic utilities to assess the site level risk. This objective concerns with safety goals, risk criteria and PRA applications for a multi-unit site. The second objective of the project is to develop methods to assess risk for multi-unit scenarios. This objective concerns with methods to identify, analyse and model dependencies between the units. In this respect, the project will go in details of the approach outlined in (Tyrräinen et al. 2017) and test the approach through pilot studies. International activities in this topic will be followed (WGRISK, IAEA).

The following subtasks are considered:

1. Risk metrics, safety goals and results presentation for site risk analysis
2. PRA methods
3. PRA model management
4. Pilot studies
5. Dissemination
6. Management, meetings
The leader of the task is Jan-Erik Holmberg. In 2017, the volume of the task will be 4.3 person months (Risk Pilot’s and VTT’s share of SITRON). Other partner of SITRON is Lloyd’s Register.

2.3.2 Dynamic flowgraph methodology (T5.2)

Dynamic flowgraph methodology (DFM) analyses systems with time-dependencies and feedback loops. As in fault tree analysis, the aim of DFM is to identify which conditions can cause a top event. The reason for the development of DFM is that traditional methods, such as fault tree analysis, can describe the system’s dynamic behaviour only in a limited manner. DFM is typically used to model and analyse digitally controlled systems that include both hardware and software components. DFM supports the modelling of multi-state components and variables, which is an advantage in modelling components with multiple failure modes and the effects that failures have on process variables. Another advantage of DFM is that only one model is needed to represent the complete behaviour of a system and therefore different states of the system can be analysed using the same model. DFM has been highlighted as one of the promising dynamic reliability analysis approaches in (Aldemir et al. 2006). In 2016, a literature study of DFM was performed (Tyrväinen 2017). The focus was especially on the applications of DFM.

VTT has developed DFM tool Yadrat in the previous SAFIR programme. Yadrat analyses DFM models by transforming them into binary decision diagrams (BDDs) (Björkman 2013). Using this approach, Yadrat is not able to solve large models. In 2017, the plan is to integrate Yadrat to FinPSA so that Yadrat models can be transformed into fault trees and solved using FinPSA’s fault tree algorithms. This way the aim is to increase Yadrat’s computational capacity and decrease computation times.

Transforming multi-state and time-dependent logic of DFM into fault trees is not straightforward, but the existing BDD implementation can be followed to some extent. A way to take dynamic constraints into account in FinPSA needs to be developed. It is likely that the prime implicants generated by FinPSA require some post-processing for which new algorithm is needed. An inconvenience is that the Yadrat source code is partially outdated and not supported by the new Java versions, but it also makes it even more important to update the tool and develop it further.

An interim report on the integration will be written. The goal is to achieve a preliminary implementation of the integration in 2017 and complete the work in 2018.

The leader of the task is Tero Tyrväinen. In 2017, the volume of the task is 1 person month.

2.4 Level 2 PRA (WP6)

This work package focuses on method development and integrated deterministic and probabilistic safety assessment (IDPSA) for level 2. On the one hand, case studies and method development are conducted. On the other hand, knowledge is transferred to assure necessary and sufficient method support for high quality modelling, analysis and result utilization in the level 2 problem domain.

<table>
<thead>
<tr>
<th>Partners in WP6</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT senior scientist</td>
<td>0.4</td>
</tr>
<tr>
<td>VTT research scientist</td>
<td>9.3</td>
</tr>
</tbody>
</table>

2.4.1 IDPSA (T6.1)

In 2015, a summary on previous IDPSA research at VTT was prepared (Tyrväinen 2015). It covered case studies on steam explosions, passive autocatalytic hydrogen recombinators, passive containment cooling system and ex-vessel coolability. In the studies, both probabilistic and deterministic analyses were performed. Especially, it was demonstrated how the results of deterministic simulations can be incorporated into a containment event tree (CET) model.
During 2016, release heights, release energies and hydrogen explosions have been studied based on literature (Tyrväinen and Karanta 2017). Full uncertainty analysis has also been developed for a simplified BWR CET model. A conference paper related to this model was published in PSAM 13 conference (Tyrväinen, Silvonen and Mätäsniemi 2016).

In 2017, deterministic analyses performed in 2016 in the CASA project will be reviewed, and the results will be utilised in CET modelling if possible. In the CASA project, deterministic analyses have been performed for steam explosions and will be performed for hydrogen explosions occurring outside BWR containment. Level 2 modelling and analysis will also be studied in the context of the integration of PRA levels 1 and 2. The focus of the task depends very much on how much input deterministic analyses can offer for PRA modelling. The utilization of deterministic analyses is the priority, but if not much can be done based on them, then the integration of PRA levels 1 and 2 will be the main focus.

The joint project SPARC, funded by NKS/NPSAG and started in 2016, will be continued with Kungliga Tekniska Högskolan (Pavel Kudinov), Lloyd’s Register Consulting (Sweden) and CASA project. The theme of the project is the scenarios and phenomena affecting risk of containment failure and release characteristics. The project involves case studies (e.g. hydrogen explosions, steam explosions, and debris bed formation and coolability). From the case studies, lessons will be drawn concerning pressure load and debris bed temperature, and from these, about the probability of a fracture in the containment.

The task will be performed by VTT. In 2017, the volume of the task will be 1.3 person months. The task leader is Tero Tyrväinen.

2.4.2 Method support for level 2 (T6.2)

Study of level 2 modelling and analysis possibilities has been started in SAFIR2014 programme. It has been noticed that IDPSA modelling approach supported by SPISA is still relevant, necessary and dominating for the level 2 needs. In addition, the new SAFIR2018 framework plan states that knowledge transfer, development and research are still needed in this area. To maintain knowledge included in these models and to provide new analysis possibilities the renewal of method support has already been started. In the present programme, FinPSA code has been developed further. Currently, the first release of the new level 2 tool is ready and has been evaluated by students, and an implementation of tight integration between PRA levels 1 and 2 is ready. The planned work of this task will provide new features for level 2 method support and provide possibilities for knowledge transfer in deep level.

In 2015, the implementation of a risk integrator to combine analysis results from multiple containment event trees (CET) was carried out. In addition, integration with the existing statistical analysator was made in algorithm and partly in implementation level. This means that statistical analysator and risk integrator utilize same algorithms for statistical parameters, percentiles and result visualization. Error reporting was improved to provide informative guidance and error messages during model building and analysis. Also the quality and overall usability of the tool was improved to better survive in error and abnormal situations e.g. runtime errors such as division by 0. Additionally, tests were planned, executed and reported for the previous actions. In 2016, the task had no activities.

The following topics will be concerned in year 2017:

- Monitoring support is needed for level 2 model analysis. Concretely, this means variable viewer updates to monitor values of variables and the most important level 1 sequences during level 2 analysis. Also, functions to control computation are designed and implemented in order to provide goal oriented model solving for PRA analyzers.

The following items are possible topics in 2018:

- To improve modelling and to avoid errors while defining system behaviour in severe accidents explicit guidelines can give immediate feedback for a modeller. Some kind of modelling language aware or syntax directed modelling features can be implemented. This kind of modelling editor enables also better traceability and visualization of runtime errors.

- To support model validation and traceability in offline the needed metainformation could be specified, its granularity could be studied and information could be embedded into model structures. Metainformation
can contain e.g. time instants with user actions and information of log files. In addition, features to handle and analyse this kind of metainformation could be designed and implemented. To improve the support of validation, traceability of model structures and computation results at different stages intermediate presentation analysators could be designed and implemented.

- Level 2 analysis results are needed in reporting, authority co-operation and with risk informed applications. The purpose of this topic is to collect reporting and integration requirements. In addition, to concretise requirements and to support further development a couple of use cases for reporting or co-operation could be specified.
- To support data exchange procedures according to specified use cases the needed data formats and exchange technologies could be defined. Also, preliminary support for procedures could be demonstrated.
- To assure high quality in tool changes the previous topics require also minor actions in level 2 test plan, execution and reporting.

The task will be performed by VTT. In 2017, the volume of the task will be 1.0 person months. The task leader is Teemu Mätäsniemi.

2.4.3 Method support for levels 1 and 2 tight integration (T6.3)

Method support for tight integration of PRA levels 1 and 2 was developed in 2016. Thus, the contributions of level 1 accident sequences and basic events to source terms and release categories can be calculated. The approach is based on idea that level 1 sequence (minimal cut set) information is carried from level 1 to level 2 and is taken into account in computation. However, PRA models are coming wider and more coupled to arise performance issues in model development, analysis and software validation. Also, more information is generated into results. These aspects slow down the development and maintenance of living PRA models, PRA documenting and reporting in plant operation organizations and review analysis of authority. Fully functional tight integration enables efficient handling of level 1 uncertainties in level 2 analysis. In addition, severe accident phenomena and dependencies can be modelled wider and more multifaceted in PRA. The approach helps the evaluation of systems and operations used in severe accident management.

To provide fully functional method support and results in better form the following topics will be concerned in year 2017:

- Analysis performance will be updated. Possible solutions are parallel analysis, deterministic result sampling and automated analysis. Primary goal is to implement parallel analysis to support analysis of multiple CETs at same time. In addition, the task educates experts to understand parallel execution implementation of the level 1.
- The solution requires also changes to the features of modeller, solver, statistical analysator and risk integrator. More control is needed in order to orchestrate co-operation of these modules. Implementation changes of the risk integrator are the most challenging because the risk integrator integrates results of several autonomous CETs. However, the topic has a good potential to make analysis performance better.
- In addition, minor changes to the user guide are also needed to inform PRA analysers.
- To assure high quality in the method support the previous topics require actions in level 2 test plan, execution and reporting. Test cases, executions, and test report will be updated.

The task will be performed by VTT. In 2017, the volume of the task will be 7.4 person months. The task leader is Teemu Mätäsniemi.

2.5 Level 3 PRA (WP7)

This workpackage consists of case studies and method development, level 3 software development, and integration of levels 2 and 3.

The case studies and method development aim to shed light on important issues in level 3, demonstrate the application of different methods to level 3 problems, and develop improved methods for level 3 analyses. These activities will be conducted so that they will also aid software development in level 3. The case studies will be
selected for each year based on discussions with Finnish level 3 experts, and also on the needs of software development. Methodological issues to be handled will also be selected on these grounds, and may include e.g. uncertainty assessment, level 3 risk importance measures, aquatic dispersion, and evacuation modelling and analysis.

The impact of external events to nuclear accident consequences will be considered. An external event may have an impact on consequences because it has been the initiating event that led to the core damage and release, or because it affects circumstances in the areas affected by the accident consequences. Taking external events systematically into account in level 3 PSA analyses would improve the accuracy of analyses and credibility of analysis results, and support improvements in emergency planning. In the first year, qualitative methodology for the purpose will be considered; in later years, quantitative methodology may be developed and case studies may be carried out.

Integration of level 2 and 3 will be considered in later years. The task will aim at a clearer understanding of what information level 3 analyses require from level 2, and how to assess the impacts and importance of different initiating events, basic events and accident sequences at level 3. It will start with the identification of a suitable case. The needed information will be mapped through the analysis needs identified in the study. State of the art will be reviewed concerning what level 3 analyses presuppose from levels 1 and 2. The role of IDPSA in level 3 modelling and analysis will be clarified, as well as its information needs. Methods will be developed to assess the effects of individual accident sequences to health and economy. One goal is also to extend the traceability of level 3 consequences back to accident sequences on level 1, or to even fractions of sequences.

2.5.1 Method development and case studies (T7.1)

In 2015, research was carried out domestically and in Nordic cooperation. A case study of the Fukushima accident was carried out, with the aim of seeing whether more sophisticated analyses bring substantial changes to the results of (Karanta et al. 2014). The most important changes were made to the modelling of wind speed (now handled as a continuous variable), and in uncertainty analysis where the uncertainty in the source term was taken into account. It turned out that these changes did not significantly change the number of estimated cancer deaths: it remained low. Nordic cooperation with ÅF Consult, Vattenfall, Risk Pilot, and Lloyd’s Register from Sweden continued in the NKS-funded project “addressing off-site consequences”. In 2015, the project concentrated on writing a guidance document for level 3 analyses.

In 2016, research has been carried out on two fronts. On the one hand, literature on population dose estimation has been carried out, emphasizing method used recently (in Fukushima and in SOARCA), and methods used in the Nordic countries. A conference paper on previous studies related to level 3 PSA has been written and presented in the PSAM13 conference.

On the other hand, Nordic cooperation with ÅF Consult, Vattenfall, Risk Pilot, and Lloyd’s Register from Sweden is carried out in the NKS-funded project “addressing off-site consequences”. In 2016, the guidance document for level 3 analyses produced in the project has been improved based on feedback mainly from Swedish stakeholders. Also a conference paper to the PSAM13 conference, concerning the guidance, was written and presented.

In 2017, the impact of initiating events and external circumstances on level 3 analyses will be considered. Initiating events, produced in level 1 PSA analyses, may have large influence on how analyses on level 3 should be conducted. For example, the initiating event of the Fukushima Daiichi nuclear accident, tsunami, caused that large areas around the power plant were depopulated before the start of the release due to death and evacuation; it is therefore reasonable to state that tsunami as an initiating event may change evacuation calculations, population dose estimation, and its results quite radically. As another example, if the initiating event is that ice blocks water intake, then obviously only weather data from wintertime should be used in atmospheric dispersion computations. Factors that affect the impact of the initiating event on level 3 will be considered: a prime example of this is the effect of time delays in accident progression, which has an effect on how weather phenomena as initiating events should be handled. External events may also have impact on level 3 analyses even when they are not initiating events; for example, very bad weather may hamper evacuation actions. Also seasonal variation has effects on level 3 analyses: for example, snow generally reduces the dose received from groundshine. Not all external circumstances will be considered; for example, ordinary weather phenomena are taken into account in current level 3 analyses. Therefore there is need to consider only rare phenomena and circumstances, for example freezing rains, that may have major impacts to nuclear accident consequences if they occur during the accident.

A systematic qualitative analysis of the potential impact of initiating events and external events will be carried out. The objective of the study is to shed light on the question: how should initiating events, seasonal variations
and rare circumstances (e.g. rare weather phenomena) be taken into account in level 3 PSA input data, models, analyses, and interpretation of results. Potential external events will be vetted through, and assessments made on how they might affect circumstances during or after a radionuclide release so that they should be taken into account in level 3 PSA analyses. Also an effort is made to conceptualize how they should be taken into account in level 3 PSA, clarifying their impact and induced dependences on level 3 elements (atmospheric and aquatic dispersion, dose assessment, countermeasures). Requirements for data needed in taking external events into account will be specified. The list of potential external events will be screened so that only events that may have substantial impact on level 3 PSA will remain on the list. Both screening and analyses will be conducted emphasizing the Finnish (or Nordic) point of view. A framework for taking external events systematically into account in level 3 PSA will be outlined.

The task will be performed by VTT. The task leader is Ilkka Karanta.

<table>
<thead>
<tr>
<th>Partners in task 7.1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT senior scientist</td>
<td>0.9</td>
</tr>
</tbody>
</table>

2.5.2 Development of level 3 analysis programs (T7.2)

The task will not be implemented in 2017.

2.6 Risk analysis of organizations and operations (WP8)

Work package 8 previously consisted of two subtasks, In task 8.1, performed by VTT, schedule risk analysis of emergency operations was considered in 2015. Schedule risk was defined as the probability that the operation or a specified part of it is not completed in a given time frame. This probability was used to define PRA importance measures (Fussell-Vesely, risk reduction worth etc.) in the context of schedule risks in resource-constrained emergency operations. It was shown how to interpret importance measure concepts in this context. Various kinds of events that may cause delay in operation completion were defined. It was also shown how to estimate importance measures from simulated, experimental or observational data. The importance measures were applied to an operation of clearing roads leading to a nuclear power plant from fallen trees after a storm. A stochastic activity network model developed in the PRADA project was used and improved, numerical estimation of the importance measures was implemented, and numerical estimates of importance measures were presented. Task 8.1 will not be implemented in 2017.

The other task, T8.2, is described below.

2.6.1 Reliability analysis of defence-in-depth in organizations (T8.2)

In SAFIR2018, this task addresses two topics in relation to defence-in-depth strategies: (i) the assessment of the impact of communication and coordination errors on organizational decision-making processes and the subsequent implementation of activities to support safety goals; and (ii) the development of robust strategies for ensuring the safety of systems, including contexts where there is a need to detect and mitigate risks through cost-effective targeting of inspections.

In 2015, the research has focused on topic (ii). The results have been reported in a refereed conference paper which has been presented at ESREL 2015 (Compare et al., 2015) and in a paper on risk-based optimization of pipe inspections in large networks on the basis of imprecise information (Mancuso et al., 2016). The research has also lead to the formulation of a PRA framework in which defence-in-depth strategies can be evaluated by using Bayesian networks as a modelling approach.

T8.2 has not received VYR funding in 2016. Consequently the plans to extend the PRA-framework toward fire safety have not been pursued. On the other hand, the development of an optimization framework, based on the conversion of standard fault trees into Bayesian networks, for the purpose of identifying effective defence-in-depth strategies has resulted in a promising methodology for which illustrative numerical studies in the area of nuclear
safety have been developed. This methodology has been presented at the ESREL 2016 conferences and pre-

sented in detail in a paper which has been submitted for publication.

In 2017, the main research objectives of the project at Aalto are to

i. develop computationally efficient algorithms for the Bayesian portfolio optimization for solving portfolio optimization problems; possibilities for implementing these algorithms into the FinPSA software will be explored;

ii. design interactive approaches in which the user can guide the selection of the risk management actions that are covered simultaneously in the portfolio optimization problem;

iii. develop methods and algorithms for handling incomplete probability information in the Bayesian portfolio optimization approach which has been formulated in the project.

The project will address the last objective also in view of assessing how possible errors in probability estimates impact the reliability of the overall results concerning system reliability, obtained after these estimates have been employed as one of the inputs for the selection of risk management actions. The close collaboration with Prof. Enrico Zio and Dr. Michele Compare (Politecnico di Milano) will continue, and collaborative activities with an industrial partner will be sought.

The task leader is Professor Ahti Salo.

<table>
<thead>
<tr>
<th>Partners in task 8.2</th>
<th>Person months</th>
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<tbody>
<tr>
<td>Aalto University professor</td>
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<tr>
<td>Aalto University doctoral student</td>
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<tr>
<td>Risk Pilot senior expert</td>
<td>0.1</td>
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</table>
## 3. Deliverables 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D2.3.1</td>
<td>Conference paper on HRA for advanced control rooms. Acceptance criterion: the paper has been accepted and presented in the PSAM Topical Conference on Human Reliability, Quantitative Human Factors and Risk Management.</td>
<td>1.5</td>
<td>7.6.2017</td>
</tr>
<tr>
<td>D2.3.2</td>
<td>Interim report or submitted journal manuscript on HRA for advanced control rooms</td>
<td>2.8</td>
<td>31.1.2018</td>
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<tr>
<td>D2.4.1</td>
<td>Travel report(s) on the participation in IAEA meetings for the preparation of the Safety report on Human Reliability Assessment for Nuclear Installations</td>
<td>1</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D5.1.1</td>
<td>Work report on site level risk metrics</td>
<td>0.5</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D5.1.2</td>
<td>Work report on site level PRA methods with emphasis on level 1 PSA for reactor units. Acceptance criterion: the report has been written and accepted by reviewers.</td>
<td>1</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D5.1.3</td>
<td>Interim work report on site level PRA model management</td>
<td>0.5</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D5.1.4</td>
<td>Interim work report summarising status of the case studies related to site level PRA</td>
<td>1</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D5.1.5</td>
<td>SITRON project meeting/workshop notes (two meetings planned with stakeholders in 2017)</td>
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<td>31.12.2017</td>
</tr>
<tr>
<td>D5.2.1</td>
<td>Report on transforming DFM models of Yadrat into fault trees of FinPSA</td>
<td>1</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D6.1.1</td>
<td>Report on level 2 PRA modelling of severe accident phe-</td>
<td>1.3</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D6.2.1</td>
<td>Updated user guide to include implemented features of variable viewer.</td>
<td>1.0</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>---</td>
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<td>---</td>
</tr>
<tr>
<td>D6.3.1</td>
<td>Demonstration of level 1 and 2 parallel analysis. The program is presented and run, and the results of the program are shown. Acceptance criterion: the demonstration shows how parallel execution runs and results are gathered.</td>
<td>5.0</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D6.3.2</td>
<td>Test plan and report covering features of parallel analysis</td>
<td>2.4</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D7.1.1</td>
<td>Report on the impacts of external events and seasonal effects to level 3 PSA, either as initiating external events or as occurring during or after the release. The report describes qualitatively what impacts external events may have on accident consequences, and how they could be taken into account in level 3 PSA. Acceptance criterion: the report has been written and accepted by reviewer(s).</td>
<td>0.8</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D71.2</td>
<td>Guidance document on level 3 PSA for Nordic conditions</td>
<td>0.1</td>
<td>31.3.2017</td>
</tr>
<tr>
<td>D8.2.1</td>
<td>Report(s) on (i) efficient algorithms for solving portfolio optimization problems in Bayesian networks, (ii) interactive approaches which allow users to guide the selection of risk management actions in portfolio optimization, and (iii) modelling techniques for handling incomplete probability information in PRA models for defense-in-depth. Reports are papers which have been accepted after refereeing for presentation at scientific conferences or submitted to scientific journals. Acceptance criterion: two or more papers have been accepted for presentation in a scientific conference or submitted to a refereed scientific journal.</td>
<td>3.5</td>
<td>Papers due by 30.9.2017 and 31.1.2018</td>
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<td><strong>Total pm</strong></td>
<td><strong>23.2</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
4. Project organisation

The project is organized as follows:

- the project manager will be Ilkka Karanta (VTT, Risk management team). The deputy project manager is Teemu Mätäsniemi (VTT).
- the organisation responsible for the whole project is VTT (manager of organization-specific work senior scientist Ilkka Karanta). Other participating organizations are Risk Pilot (office manager Jan-Erik Holmberg) and Aalto University (professor Ahti Salo).
- the partners in joint activities are as follows: Risk Pilot (topical issues in HRA); Aalto University (risk analysis of organizations)

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stefan Authén</td>
<td>Senior consultant</td>
<td>Risk Pilot</td>
<td>T5.1</td>
<td>1.0</td>
</tr>
<tr>
<td>Kim Björkman</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T5.1, T5.2, T6.2, T6.3</td>
<td>3.45</td>
</tr>
<tr>
<td>Jan-Erik Holmberg</td>
<td>Office manager</td>
<td>Risk Pilot</td>
<td>T2.4, T5.1</td>
<td>2</td>
</tr>
<tr>
<td>Carl Sunde</td>
<td>Consultant</td>
<td>Risk Pilot</td>
<td>T5.1</td>
<td>0.6</td>
</tr>
<tr>
<td>Maria Frisk</td>
<td>Consultant</td>
<td>Risk Pilot</td>
<td>T5.1</td>
<td>0.6</td>
</tr>
<tr>
<td>Ilkka Karanta</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T1, T6.1, T7.1</td>
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</tr>
<tr>
<td>Terhi Kling</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T2.3</td>
<td>1.5</td>
</tr>
<tr>
<td>Marja Liinasuo</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T2.3</td>
<td>0.7</td>
</tr>
<tr>
<td>Teemu Mätäsniemi</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1, T6.2, T6.3</td>
<td>3.4</td>
</tr>
<tr>
<td>Alessandro Mancuso</td>
<td>Trainee</td>
<td>Aalto University</td>
<td>T8.2</td>
<td>8</td>
</tr>
<tr>
<td>Markus Porthin</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T2.3</td>
<td>2.1</td>
</tr>
<tr>
<td>Ahti Salo</td>
<td>Professor</td>
<td>Aalto University</td>
<td>T8.2</td>
<td>0.5</td>
</tr>
<tr>
<td>Carl Sunde</td>
<td>Senior consultant</td>
<td>Risk Pilot</td>
<td>T5.1</td>
<td>1.0</td>
</tr>
<tr>
<td>Tero Tyrväinen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T5.1, T5.2, T6.1, T6.3</td>
<td>4.5</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>31.65</strong></td>
</tr>
</tbody>
</table>

The project has connections to some other projects within SAFIR2018

- Integrated Safety Assessment and Justification of Nuclear Power Plant Automation (SAUNA, Antti Pakonen / VTT) project aims at developing means for safety demonstration with a focus on process monitoring and control. PRAMEA and SAUNA share the themes of safety assessment and defence-in-depth. Some researchers work in both projects.
- Extreme Weather and Nuclear Power Plants (EXWE, Kirsti Jylhä / Finnish Meteorological Institute) aims at supporting overall safety of nuclear power plants by enhancing scientific understanding of the environmental conditions of the plant's location and predicting how they can change. Extreme weather is an important reason of initiating events in PSA level 1, and weather conditions are central in the atmospheric dispersion of release in PSA level 3.
- Fire Risk Evaluation and Defence-in-Depth (FIRED, Anna Matala / VTT) aims at doing fire research that supports risk analysis. Fire is also an important initiating event in PSA level 1, and fire research may produce results relevant to level 2 analyses, too.
- Analysis of Fatigue and Other Cumulative ageing to exteND lifetime (FOUND, Juha Kuutti / VTT) concerns cross-disciplinary assessment of ageing mechanisms for safe management and extension of
operational plant lifetime. The work in the project includes probabilistic structural safety assessment of NPP piping systems and development of risk-informed in-service inspection (RI-ISI) methodologies. There is a task for connection between PRA and RI-ISI analyses, where a new RI-ISI feature has been developed for FinPSA. One research scientist works in both projects.

- Comprehensive Analysis of Severe Accidents (CASA, Anna Nieminen/VTT) aims at developing improved safety analysis methods for severe accidents and consequence analysis. These are closely related to PSA levels 2 and 3. PRAMEA and CASA collaborate mainly through participation in SPARC project, concerning scenarios and phenomena affecting risk of containment failure and release characteristics, and partially funded by NKS.

The project will have collaboration with foreign partners in the following international activities:

- Markus Porthin is a member of OECD/NEA WGRISK and ETSON, and will participate in their meetings and groupwork.
- In WP2 (HRA), Jan-Erik Holmberg will participate in the IAEA work group to prepare a safety report on HRA.
- In task 5.1 (site-level PRA) there will be cooperation with Lloyd’s Register Consulting (Sweden) concerning the development of site risk analysis methods together with support from the Swedish stakeholders Ringhals Ab, Forsmarks Kraftgrupp and SSM.
- Task 6.1 (integrated deterministic and probabilistic risk assessment) is a part of an NKS project called SPARC. The project partners are Kungliga Tekniska Högskolan (Sweden), Lloyd’s Register Consulting (Sweden) and CASA project.
- In task 7.1, there will be collaboration with Swedish partners (Lloyd’s Register, ÅF Consult, Risk Pilot, Vattenfall) in finalizing the guidance document developed in previous year, and in having a workshop on level 3 PRA.

A project called **New Approach to Reactor Safety ImprovementS** (NARSIS) has been resubmitted to the Horizon 2020, EU Framework Programme for Research and Innovation. The project has three goals: improving the characterization of natural external hazards and considering concomitant external events, either simultaneous-yet-independent hazards or cascading events; improving the vulnerability assessment of the elements subjected to complex hazards and introducing a vector-based approach for fragility representation; and improving the integration of risks and the treatment of uncertainties including those related to the integration of the expert judgment in the PSA. The project has 17 partners from 9 countries; VTT is among the project partners.
5. Risk management

The following risks have been identified. They are listed together with respective mitigation actions.

- key research objectives may not be reached if there are problems with human resources. To mitigate this, the reference group can modify the plans and budget in accordance to external pressures on resourcing. The consequences of the loss of an expert will be mitigated by keeping more than one expert aware of progress in each task; this will reduce the risk that knowledge will disappear simultaneously with an expert. Funding may be reallocated within the project to prevent compromising key research objects.
- there may be difficulties in coordination and cooperation, because the project group consists of individuals from many research fields and organizations, and most are working together for the first time. This will be mitigated by specifying clear objectives for each task. There will also be monthly meetings where progress is monitored within the project group. A cooperation agreement will be made between VTT and other organizations (Risk Pilot and Aalto University), and it specifies the responsibilities of each organization.
- VYR funding may be cut, if the Hanhikivi 1 project does not proceed as planned. If this happens, the project plans will have to be revised and objectives of the project will have to be scaled down accordingly.

VTT’s operational system has been certified in accordance with the ISO 9001:2008 standard. The certificate was issued by DNV in 2006, and covers research, technology transfer and consultation services and the development of new technology at VTT.

Risk Pilot’s work is performed in accordance with Risk Pilot’s internal quality assurance system which is in accordance with ISO 9001.
References


### PRAMEA

**Total budget (VTT + Aalto + RiskPilot)**

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Material</td>
<td>Personnel</td>
</tr>
<tr>
<td>VTT</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Aalto University</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Risk Pilot's budget</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Comments:

| Other = Helander in-kind, Ringhals AB, Forsmarka Kraftgrupp AB |
| For T5.1 only. VTT's and RiskPilot's contribution is shown. The STTROXproject has also Lloyd's Register Consulting as a partner. For Swedish funding, NKS 2017 currency rate is used. |

### VTT budget

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
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</thead>
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<td>Personnel</td>
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<tr>
<td>Aalto University</td>
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<td></td>
</tr>
<tr>
<td>Risk Pilot's budget</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Comments:

| Other = Helander in-kind, reduced funding |
| Travel: T1.1 Etson meeting, OECD WGRISK; T5.1, T6.1, T7.1 Nordic cooperation meetings (Norway/Sweden), T2.3 EHPG, Nordic cooperation meetings (Norway/Sweden), T6.1 PSA |

### Aalto University budget

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
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</thead>
<tbody>
<tr>
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<tr>
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<tr>
<td>Aalto University</td>
<td></td>
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</tbody>
</table>

| Other = Ringhals AB, Forsmarka Kraftgrupp AB = 220 kSEK |

### Risk Pilot's budget

<table>
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<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
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</thead>
<tbody>
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<td></td>
</tr>
<tr>
<td>Aalto University</td>
<td></td>
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</tbody>
</table>

Comments:

| Travel = trip to Vienna (AISA Safety guide on HPA) and Stockholm (STTROX) |
| Other = Ringhals AB, Forsmarka Kraftgrupp AB = 220 kSEK |
SAUNA 2017

Integrated safety assessment and justification of nuclear power plant automation

Antti Pakonen, Jarmo Alanen, Kim Björkman, Hanna Koskinen, Nikolaos Papakonstantinou, Markus Porthin, Teemu Tommila, Janne Valkonen
VTT Technical Research Centre of Finland Ltd

Valeriy Vyatkin, Igor Buzhinsky
Aalto University

Timo Varkoi, Risto Nevalainen
Finnish Software Measurement Association (FiSMA)

Jan-Erik Holmberg
Risk Pilot Ab

Eero Uusitalo, Mika Koskela
IntoWorks
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1. Research theme and motivation

Companies operating in regulated areas are obliged to provide comprehensive and documented evidence for the safety of their products and applications. The aim of the SAUNA project is to develop integrated and multidisciplinary ways to build confidence in the safety of nuclear power plants and their systems. Within the general framework of Systems Engineering (SE), this goal is pursued by developing comprehensive and transparent safety demonstration practices. The results, intended to be used by the regulator, the utilities and by the system suppliers, cover co-use of diverse assessment methods, structured representation of the safety justification arguments and common working processes for system development and licensing. While scoping the project primarily on plant operations, i.e. on instrumentation and control systems and their human users, the SAUNA team considers a power plant and organisations involved in its development, licensing and maintenance as a sociotechnical system.

1.1 Background and state-of-the-art

Nuclear power has long traditions in safety engineering. Design basis assumptions and identified initiating events are the starting point for defining the necessary safety functions and systems. Deterministic and probabilistic analysis methods are the way to show that regulatory requirements have been satisfied. However, some problems can be identified. Various disciplines tend to work in isolation according to their own traditions. Engineers focus on technical details putting less emphasis on users’ needs, requirements and system functions. Terminologies and practices differ among countries and organisations. Interpretation of regulatory requirements and standards is often experienced as a challenge. Document-based design leads to difficulties in information management and communication in the supplier-utility-regulator loop. Deficiencies in safety culture may lead to neglect of important safety issues and making of false assumptions during the design process. These challenges are similar to other safety-critical domains, and as often admitted, nuclear power might benefit from best practices in other application areas.

A new challenge of the nuclear industry is related to the risks associated with extreme external events and complex failure combinations, such as the problems encountered during the Fukushima accident. These rare but severe situations are beyond conventional safety design practices and have given rise to lively debate about research needs within the nuclear community (e.g. SNETP 2013 and NUGENIA 2013). This highlights the importance of emergent, system-level phenomena. The traditional, technology-driven concept of Defence-in-Depth (DiD, WENRA 2013, IAEA 2005) should be given an extended meaning and its use reinforced. Allocating safety functions to plant systems and structures and to human organisations and procedures on various DiD levels is a difficult design task encountered already during the conceptual design stage (Figure 1). New methods should be developed to represent the safety architecture and to analyse its dependencies, vulnerabilities and robustness in unexpected situations. A robust design based on DiD with sizeable safety margins and diverse means for delivering the safety functions, as well as operator response plans, will help to protect against the unanticipated (WENRA 2013). Similar principles of successive lines of defence against human error should be applied also in organisational processes, e.g. in design, maintenance, management and regulatory oversight. However, the different principles of organisational behaviour (in relation to technological systems) need to be taken into account, as well as the overall – common-cause – influence of safety culture on all organisational processes.
Companies operating in regulated areas like aviation, railroads and pharmaceuticals, must provide documented evidence for the safety of their products and applications. The FP7 EURATOM project HARMONICS (2014) has listed principles for justifying safety:

1. Effective understanding of the hazards and their control should be demonstrated.
2. Intended and unintended behaviour of the technology should be understood.
3. Multiple and complex interactions between technical systems and also human systems to create adverse consequences should be recognised.
4. Active challenge should be part of decision making throughout the organisation. Needs of all stakeholders to understand and challenge case should be taken into account in its structure and presentation.
5. Lessons learned from internal and external sources should be incorporated.
6. Justification should be logical, coherent, traceable, accessible, and repeatable with a rigour commensurate with the degree of trust required of the system.

In the tradition of the nuclear field, preliminary and final Safety Analysis Reports (SAR) support the application for authorisation during different steps of the licensing process (IAEA 2010, YVL A.1 2013). Licensing also covers the Operational Limits and Conditions (OLC) in the form of controls, limits, conditions, rules and required actions that are formally derived from the safe operating envelope. Moreover, early communication and tracking of open issues is an essential part of the licensing process. The current practices applied in Safety Analysis Reports, other design documentation and quality management are not fully satisfactory. For example, requirements specifications, traceability information, configuration management and safety argumentation are not always clearly explicated (Tommila, Savioja & Valkonen 2014).

In recent years, more clarity and rigor have been demanded from the nuclear licence applicants. More or less as a synonym for safety case, the term “safety demonstration” is used for structured arguments and evidence that support the claims on the safety of a system important to safety (Common position 2014). Safety case is not produced only for the regulator but one of its main users is the licensee itself (HSE 2006). Similar demands have been increasing also in other safety-critical areas, including long-term repositories of nuclear waste (Rasilainen et al. 2013). This has led to the emergence of standards and tools for developing demonstrations of safety, security and other properties related to system dependability. Depending on the type of top-level claims, terms like “safety

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case”, “dependability case”, etc. are used. To combine these applications of the approach, standardisation bodies and industrial consortia are developing methods, information models and tools for the generic “assurance case” and its applications (ISO 15026-1, IEC 62741 2014, Campara 2010). Also conferences have been organised, such as ASSURE2014 (see http://ti.arc.nasa.gov/events/assure2014). As illustrated in Figure 2, the structured assurance case would provide a framework for a consistent, transparent and multidisciplinary safety demonstration also for nuclear power plants and their individual systems. For more information on safety demonstrations, see (Tomilla, Savioja & Valkonen 2014) and (HWR-1112 2014).

Figure 2. Types of evidences for Assurance Case (Campara 2010).

Various assessment methods are needed to collect the evidences for an assurance case. Figure 3 shows the emergence of some well-known methods used to address technical, human factors, and organisational issues. It is noteworthy that human factors methods came onto the scene after the accident at Three Miles Island in 1979, and that organisational methods were developed following the Chernobyl and Challenger accidents in 1986. Established ways of thinking about accidents, such as the “Swiss Cheese” analogy of holes in safety barriers that ‘line up’, are deemed to be unable to prevent, predict, and explain new types of accidents (EUROCONTROL 2009; Leveson 2004). It can be expected that systemic design and assessment methods like STAMP and STPA (Leveson 2011) will gain even more importance after the Fukushima accident (see also Hollnagel 2004 about the development of accident models).

Figure 3. Accident Analysis and Risk Assessment Methods (EUROCONTROL 2009).
Nuclear power has long traditions in using several safety analysis methods. In particular, Probabilistic Risk Assessment (PRA) is important for identifying systems and events having the most impact on safety. Some methods have also been developed in the SAFiR programmes. As an example, model checking has been one of the success stories of SAFiR, and has been put to practical use in the industry with good results (Pakonen et al. 2014). Still, as the use of such state-of-the-art V&V methods calls for ad hoc solutions and manual expert work, they are not yet a part of everyday industry practice. Dedicated tools and work processes are still needed, as well as integration to the tools used for requirement specification and design.

1.2 Objectives and expected results

The fundamental challenge in nuclear power plant engineering is to ensure and demonstrate the safety of the complex sociotechnical system by considering all types of hazards including rare and extreme conditions (internal, external), all disciplines and all types of system elements (technical, human, environmental, ...) and all life-cycle phases and activities. Therefore, the overall objective of SAUNA is an integrated framework for safety assessment and transparent safety demonstration. Both the regulator and utilities, as well as system suppliers and contractors, would benefit from cost-effective and timely licensing and implementation of investments in new builds and upgrades. The SAUNA project aims to contribute to this goal in the following ways:

- On the level of fundamental safety principles, SAUNA reviews recent trends in regulatory policies, standardisation and research on the design of complex, safety-critical systems for a better understanding of various aspects of the overall nuclear power plant safety. The resulting literature reviews, roadmaps and conference papers provide insights to necessary improvements in national design and regulatory practices and to the needs for further research, especially within the SAFiR2018 programme.

- For practical design and licensing work, SAUNA takes the good practices of Systems Engineering and project management as the starting point and adapts them to the needs of the nuclear domain. Progressing from the clarification of current terminology towards more formal information models SAUNA promotes future model and database oriented design practices as a complement to traditional documents. The expected results on this level include shared reference models for various systems engineering and regulatory processes as well as documents and data models for expressing the information exchanged between the parties involved in design and licensing. Process assessment methods are developed to evaluate the quality of the systems engineering processes.

- Within the framework of Systems Engineering and principles of nuclear safety, the SAUNA project focuses on safety demonstration practices considering both the ways to represent the safety claims, arguments and evidences and the processes of safety case development and licensing. Safety assessment and justification is, however, seen together with the design activities that produce a significant part of the claims and evidences needed for safety demonstration. The results, primarily in the form of guidance for safety demonstration development and licensing, can be used during new build and upgrade projects and for periodic safety reviews of existing plants. The structured representation and development approach are expected to make the design and licensing of power plant automation smoother and more cost-effective, both for the regulator and the utilities.

- The claims presented in a safety demonstration are typically derived from regulations, standards, specific system requirements and other points of reference like “best practices in the domain”. For collecting a comprehensive set of evidences assessors need a multitude of analysis methods. The objective of SAUNA is to enhance and integrate existing methods and to develop new assessment methods where needed for a good coverage in the safety case. While each method focuses on its specific aspect of a system or its life-cycle, they all should provide useful inputs to the safety demonstration and, where practical, apply the same claim-argument-evidence logic in their working process and documentation of the observations. The results are documented in the form of research reports and manuals that can be used by the developers themselves and by independent assessors.

- The production and management of all the information needed for design and licensing is not possible if performed manually. Therefore, SAUNA also intends to develop software tools where the analysis of practical needs and implementation options make it reasonable. Examples of potential areas for tool development are management and exchange of design data (e.g. requirements, traceability, system configuration and versions, PRA), analysis methods (process assessment, model checking), and utility-regulator communication (e.g. issue management systems).
In order to limit the scope and the required efforts SAUNA focuses on design and licensing issues related to power plant operations in normal, low-power and accident situations. This means that I&C and information systems, control room(s), human operators and emergency support personnel are in the centre (Figure 4). With this focus, the project has a multidisciplinary character with links to process systems, electrical systems, spaces and structures, etc. Therefore, the results can have a wider applicability, and SAUNA wishes to exchange ideas with other research activities, for example in the SAFIR2018 and KYT2018 programmes.

Figure 4. Within the framework of systems engineering and nuclear safety, SAUNA develops means for safety demonstration with the focus on process monitoring and control.

1.3 Exploitation of the results

Table 1. Main results of the SAUNA project.

<table>
<thead>
<tr>
<th>Result</th>
<th>End user types</th>
<th>Project year(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Regulator</td>
<td>Utility</td>
</tr>
<tr>
<td>Overall reference model of the NPP Systems Engineering process</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Reference model of the licensing process</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Guidance for integrated Human Factors Engineering as part of Systems Engineering</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Description and analysis methods of plant-level DiD architecture</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Safety culture as an extension of defence in depth concept</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Guidance for probabilistic safety assessment of the plant- level DiD architecture</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Assessment method of requirements specifications</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Integrated toolset for model checking</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>Systems Engineering process assessment method</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Analysis method for the I&amp;C architecture</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Guidance for safety case development and documentation</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Human factors safety case</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Support for safety demonstration development and licensing</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

An overview of the objectives and results was given in the section above. Corresponding to the levels of focus and concreteness, the development activities are organised into three work packages in chapter 2, first one devoted to
general systems engineering principles, second to assessment methods and tools for evidence collection and, finally, third to developing and documenting the safety demonstration as part of the licensing process. The types of expected results include research and technology insights, Systems Engineering and assessment methods and tools, and guidance or safety demonstrations. They are intended to be used by the regulator, utilities and their collaborators, as well as by the research organisations themselves, for the purposes of 1) general enhancements of regulatory practices; 2) training of new experts; 3) development of in-house working practices and 4) development of service products for system and process assessment. While maintaining a scientific orientation, SAUNA wishes in its specific method and tool development to provide solutions that can be applied in the daily work of the regulator and the industrial partners. Except for the first project year (see chapter 3), the particular deliverables are not known at the time of writing. However, the table above gives examples of the main results, their intended users and the times when the results will be available (typically in several versions).

1.4 Appropriateness of the project to SAFIR2018 programme

The objectives of SAUNA are related to the SAFIR2018 research area of “Plant safety and systems engineering”. In Table 2 below, we describe how the SAUNA project supports the goals of the programme as stated in the SAFIR2018 framework plan (MEE 2014).

Table 2. Linking of the objectives of SAUNA to the goals of the SAFIR2018 programme.

<table>
<thead>
<tr>
<th>SAFIR2018 themes and goals (MEE 2014)</th>
<th>SAUNA themes and goals</th>
</tr>
</thead>
<tbody>
<tr>
<td>“The overall safety is built up from the architecture of the nuclear power plants but, at the same time, it comprises operating processes during the various stages of the plant’s life cycle. In addition to understanding the actual power plant and its technical systems, it is also important to understand the operation of the organisations and the interactions between them as factors contributing to the overall safety (“organisation or organisations”). Such a holistic understanding of safety requires broad-scale and multidisciplinary research.” (Ch. 3.2.1 p. 30)</td>
<td>SAUNA covers methods and tools related to assessment of design solutions and systems engineering processes. The power plant is viewed as a sociotechnical system of systems. Researchers with a background in current SAFIR2014 reference group areas of 1) man, organisation and society, 2) automation and 8) control room, and probabilistic risk analysis will work together.</td>
</tr>
<tr>
<td>“For the overall safety of a nuclear power plant, the central areas of research are topics located at the interfaces between functions or areas of technology […] New technical solutions require new methods for assessing a system’s behaviour and safety.” (Ch 3.2.2, p. 31)</td>
<td>Interfacing between different disciplines is one of the themes in WP2, which deals with different assessment methods, including process and reliability assessment methods. STPA is an example of a new method that in particular addresses also societal issues and human actions in hazard analysis of plant systems (see T2.3).</td>
</tr>
<tr>
<td>“The research challenge is to form a framework for the assessment of overall safety in which the traditional concept of defence in depth is clarified by integrating safety maintenance structures, process systems, automation systems, their support systems and the actions of organisations and humans in a single model.” (Ch. 3.2.4, p. 33)</td>
<td>WP1 will work on the DiD architecture, WP2 on modelling and assessing the interdisciplinary connections, and WP3 on integrating the assessment results from different disciplines into a coherent safety justification.</td>
</tr>
</tbody>
</table>
### SAFIR2018 themes and goals (MEE 2014)

“Factors essential to the management of the overall safety and significant differences between the management in the different stages of the life cycle should be identified. Depending on the stage of the life cycle, some processes, competences or management models, for example, can be emphasised more than others.”

(Ch. 3.2.4.2, p. 34)

<table>
<thead>
<tr>
<th>SAUNA themes and goals</th>
</tr>
</thead>
<tbody>
<tr>
<td>In WP1, a Systems Engineering Management Plan (SEMP) for the nuclear domain will be developed. A SEMP describes, e.g., NPP life cycle stages, SE processes, concept models of engineering data, role and collaboration models, and potentially tool integration models.</td>
</tr>
</tbody>
</table>

| WP3 will utilise a model driven approach for safety demonstration, in order to, among other things, to better capture the traceability links between requirements, design artefacts and assessment results. |

| WP1 will clarify how issues such as functional isolation relate to DiD, and a reference plant model already used for PRA will be extended to cover I&C system architectures. WP2 will then look at the assessment of I&C architectures from a DiD point of view. |

### 1.5 Education of experts

SAUNA intends to collect recommendations and experiences on safety justification from other countries and application domains and make the knowledge available to all Finnish organisations. The project will also facilitate the interdisciplinary expertise of researchers by combing a range of research topics and bringing together people with different backgrounds. Researchers from current SAFIR2014 reference groups 1, 2 and 8 will take part in SAUNA, as well as researchers with little or no previous experience on the nuclear domain. Moreover, SAUNA aims at producing a fair number of scientific journal publications in as well as participating in different international conferences. The project will also facilitate participation in forums such as the working groups of OECD/NEA and IAEA.

Regarding specific theses and dissertations, the following actions are planned:

- At Aalto, a doctoral student has been employed on a research topic related to automation applications of formal verification.
- At VTT, there are several dissertations either underway or being prepared on topics related to safety and reliability assessment methods and human factors engineering. Several of the work tasks specified for SAUNA will allow for the continuation of dissertation work already begun in SAFIR2014.
- At Risk Pilot, R&D projects such as SAUNA are used for education of younger experts in the field. 1–2 younger experts will work in SAUNA during 2015–18. Further, Prof., Dr. Jan-Erik Holmberg acts as a supervisor to VTT’s doctoral students within risk analysis methods related studies.
- IntoWorks, as a startup consultancy company, builds up its skill set throughout SAFIR2018. In 2016, a Master’s Thesis student was hired thanks to participation in SAFIR2018, and further opportunities are being investigated for 2017 and 2018.
- FiSMA supports trainees by guiding thesis work in task specific topics as appropriate. In-depth knowledge and material related to international standards in systems and software engineering will be available.
2. **Work plan**

The SAUNA project is divided into three work packages each having a planned duration of four years (Figure 5). For each project year, a set of tasks will be defined with a clear scope and deliverables. While individual tasks may be short and their titles may be changed from year to year, they work on longer-lived topics within the work packages.

SAUNA will have a multidisciplinary research strategy, and it will look at plant operations in the context of the whole plant and investment project. The purpose of WP1 is to build a shared understanding of NPP challenges and recent advances in the safety of complex systems in various domains. It also sets the stage for the other work packages in terms of common Systems Engineering principles and modelling concepts. Within this framework, WP2 develops dedicated methods and tools for assessing the safety of (planned and existing) systems and their development processes. The idea of WP3 is again to tie the results together into an integrated approach to safety demonstration and licensing. Finally, project planning and progress reporting are allocated to a separate work package WP4 that also includes the necessary activities related to research collaboration, coordination and dissemination of results.

The planned 2017 tasks for the work packages WP1 through WP3 are listed in Figure 5. The work packages and their tasks planned for 2017 are described in detail in the sections below.

### Figure 5. Tasks planned for 2017 in work packages 1 to 3.

<table>
<thead>
<tr>
<th>WP1, Safety Systems Engineering</th>
<th>WP2, Analysis methods and tools</th>
<th>WP3, Safety demonstration practices</th>
</tr>
</thead>
<tbody>
<tr>
<td>SEMP Qualification processes</td>
<td>Closed loop modelling in formal verification</td>
<td>Assurance methods Guidelines</td>
</tr>
<tr>
<td>DID Architecture DID Modelling</td>
<td>T1.1 MBSE methods for architecture level Did analysis</td>
<td>PLANS</td>
</tr>
<tr>
<td>T1.3 Modelling of digital I&amp;C (DIGREL PRA model)</td>
<td>T2.1 Explicit state model checking</td>
<td>T3.1 Overall plant automation safety demonstration</td>
</tr>
<tr>
<td>I&amp;C Ref. Architecture</td>
<td>T2.2 User-friendly requirement editing for formal verification</td>
<td>Safety of control room systems</td>
</tr>
<tr>
<td>T2.3 Application of STPA in digital I&amp;C</td>
<td>Integration of model checking &amp; PRA</td>
<td></td>
</tr>
<tr>
<td></td>
<td>T2.5 Process assessment methods and tools</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th></th>
<th>2015-2016</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
</table>

### 2.1 Safety Systems Engineering (WP1)

As stated, for example, by Nancy Leveson (2004, 2011), changes have occurred in the requirements of society and in the types of man-made systems that make us question the foundations of traditional safety engineering (see also Hollnagel 2004). For example, continuous changes, new types of threats and dependencies in complex systems can exceed the limits of human designers and operators in managing unexpected situations and learning from past experiences. A new paradigm for engineering and operating these systems is needed. In regulated areas like nuclear power, there is the further question, whether a system is safe enough and how the answer can be justified in an unarguable way.

The goal of SAUNA is to integrate various disciplines and aspects of safety into a consistent safety design and demonstration framework. As a conceptual basis for this, it reviews the challenges and recent developments in
nuclear power and in the design of safety-critical systems in general. Building upon the results from the SAFIR2014 programme, WP1 focuses on clarifying conceptual models (terminology) and the principles of Systems Engineering in order to provide the common basis for various research activities, disciplines and aspects of nuclear power plant safety. For this purpose, WP1 includes tasks related to fundamental safety principles, modelling concepts and working practices. It will also take care of shared research results (e.g. demonstrations) and carry out literature reviews of new cross-cutting themes encountered during the project. Within this common framework, the tasks in the other work packages can go deeper into their specific research issues. The yearly research tasks in WP1 will be organised under the following broad topics.

**Topic: Power plant engineering data modelling**

For successful engineering projects and collaboration in the supply chain, all participants should have common and well-founded terminologies, understanding of system architectures and functions, documentation practices and information models concerning the systems under development. The purpose of tasks in this topic is to review current practices and literature and to bring in common conceptual models, typically on a rather general level. Important safety-related areas will be subject of more detailed analysis. Examples of such issues are the Defence-in-Depth (DID) architecture and traceability of design data.

**Topic: Systems Engineering processes**

In addition to sound engineering data models, success requires disciplined and shared working processes from all participants. In the second topic of WP1, research tasks are organised to provide reference models for safety systems engineering. Under the umbrella of systems thinking and systems engineering in general, certain safety related activities, for example the safety analysis and V&V, will be brought into the focus. The development of the safety demonstration, as well as the licensing activities carried out by the utilities and the regulator will be considered in WP3.

**Topic: Shared examples**

The aim in the SAUNA project is that several tasks and even other, related SAFIR projects work around same case examples in order to share ideas and add value to the demonstrations. These examples should be public and thereby available for publications and training of new experts. The main example will be the virtual plant used in the DIGREL-project (2010–14). In 2015, the example was further developed in the MODIG (Modelling of DIGital I&C) project financed by NKS, SAFIR2018 and SSM (Swedish Radiation Safety Authority).

WP1 will be coordinated by Nikolaos Papakonstantinou (VTT). Partners and person months allocated to WP1 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>7</td>
</tr>
<tr>
<td>Risk Pilot</td>
<td>1.5</td>
</tr>
</tbody>
</table>

WP1 will be coordinated by Nikolaos Papakonstantinou (VTT). Partners and person months allocated to WP1 are given in the table below.
2.1.1 Model Based Systems Engineering (MBSE) methods for architecture level DiD analysis (T1.1)

The early development phase of complex systems is critical. Design weaknesses introduced at the beginning of the plant’s design, often emerging from behaviours caused by interdisciplinary interactions, can be very costly to be addressed later on during the project’s development. During 2015, the SAUNA project delivered a comprehensive state of the art report on Defence in Depth (Tommila & Papakonstantinou 2016) clarifying the related terminology and learning about the importance of interdependencies and human organisations. During 2016, a metamodel supporting early interdisciplinary modelling and a prototype tool were developed and tested in a small case study of a spent fuel pool cooling system. The aspects included were the environment (rooms, corridors, staircases), process, automation (HMI, cabling and actuators/sensors), electrical supply, operators and a functional decomposition of the system. This meta model is complementing and can be linked to the modelling approach (SEAModelNPP) developed by Tommila & Alanen (2015) in the SAREMAN project during the SAFIR2014 programme. SEAModelNPP is a Systems Engineering artefacts model for the engineering data to facilitate traceability of design and V&V artefacts to requirements.

In 2017, the overall objective is to demonstrate the possibilities provided by structured system models and computer-assisted analysis techniques. More specifically, this task will focus on I&C systems and further development of the meta model, definition of requirements and inference rules for analysis and the further development of the case example. The metamodel will be refined on the basis of principles in the SEAModelNPP. As planned, the example of a spent fuel pool will be used and refined for early dependency and fault propagation analysis as part of Defence in Depth and I&C architecture assessment. The results of these analyses and assessments will be utilized in task T3.1 to build a safety demonstration. Extensions to the case study will also provide connections to early HRA (using the early HRA assessment past work, deliverable of SAUNA 2016) and potentially to PRA in collaboration with task T1.3.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Nikolaos Papakonstantinou (VTT)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Teemu Tommila, Joonas Linnosmaa, Jarno Alanen (VTT)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>4.3</td>
</tr>
</tbody>
</table>

2.1.2 MOdelling of DIgital I&C (MODIG) (T1.3)

MODIG follows DIGREL-project with an objective to support the use of the PRA in both deterministic and probabilistic risk evaluations of digital I&C. The task is realized through three main tiers:

- Defense in depth (DiD) analysis supported by PRA
- Software reliability modelling of digital I&C
- International collaboration.

In addition, in 2017 the subtask “integration of model checking and PRA” will be included in this task. This work focuses on developing unified safety assessment approaches for plant safety analysis based on integrating model checking and PRA. There exist several potential integration approaches (Lahtinen, & Björkman 2015, Björkman et al. 2015)

In the previous years (2015-2016) of the project, a survey of the defence-in-depth (DiD) framework and the role of PSA in the framework has been conducted (Authén et al. 2016, Holmberg et al. 2017). Regarding digital I&C, the focus has been traditionally in DiD level 3, reactor protection system. In MODIG, attention has been paid into DiD level 2 preventive safety functions which have dependences with DiD level 1 and 3 functions. Properties and requirements for DiD level 2 I&C functions have been discussed, and the DIGREL model has been expanded with a simple example of a preventive safety function. Based on the findings from the simple example further areas of improvement in the modelling has been recognised. Such areas are e.g. the better categorisation of initiating events and the consideration of spurious actuactions in system dependencies.

The software reliability modelling developed within DIGREL and MODIG has described how to separate RPS software into software entities that can be analysed and modelled separately (Bäckström et al. 2016, Tyrväinen et al. 2016). Generally, for system software operational experience is considered relevant to estimate failure data. For application software an analytical approach is needed to estimate failure data due to insufficient operational experience. The current work has identified that it is necessary to further discuss exchangeability between especially application software (how can software be considered to be the same – that is, how operational...
experience can be pooled and used as evidence). Also, the project has outlined the concept of claims and evidence to continue the justification of the method.

A technical session on PSA Experience in modelling digital I&C was held as part of the 17th meeting of OECD/NEA Working Group on Risk Assessment (WGRISK) March 2016. This topical discussion was led by U.S. NRC and VTT. As a result of the discussion, it was agreed to prepare for a new WGRISK activity in form of a benchmark study on reliability of digital I&C, to compare different modelling approaches. The preparation of the new task proposal is led Korea (KAERI), supported by Finland (VTT), USA (U.S. NRC) and Germany (GRS). It is planned to use the DIGREL PSA model as a starting point.

Regarding integration of model checking and PRA, a case study was performed in order to demonstrate the usability of the developed integrated model checking and PRA approach (Lahtinen & Björkman 2016). The main idea of the approach is that the model-checking analysis is restricted based on the PRA results.

Tasks for 2017

1. **Assessment of DiD using PRA**
   
   DiD assessment approach outlined in 2016 (Holmberg et al. 2017) is further specified with an emphasis on defining an acceptable safety case for the NPP’s safety I&C. The safety case will cover the main deterministic and probabilistic DiD requirements. The basis for the safety case is taken from PRA since it provides a systematic, comprehensive and logically sound representation of NPP’s hazards and system dependences. One aim with the proposed safety case is to highlight the most critical claims needed in the licensing of NPP’s safety I&C, e.g., completeness and representativeness of considered scenarios, justifiability of screening rules and validity of analysis tools and models.

2. **Software reliability**
   
   This subtask will not be active in 2017 due to negative funding decision by NKS.

3. **WGRISK benchmark study on reliability of digital I&C**
   
   A new task concerning a benchmark study on reliability of digital I&C is foreseen to be initiated by OECD/NEA WGRISK in 2017. MODIG will participate in the preparation of the task proposal as well as in the completion of the task. It is planned to use the DIGREL PSA model as a starting point when defining the benchmark I&C system. Decision is made in the WGRISK annual meeting (8-10th March 2017). The main part of the digital I&C PSA modelling of the benchmark example and reporting is done during 2018.

4. **Integration of model checking and PRA**
   
   A more complex case study will be developed to test the applicability of the approach on a larger scale model. The DIGREL model may serve as a large scale case study. The development of the case study will also benefit the development of an enhanced automation modelling approach for PRA (possibly addressed in the PRAMEA SAFIR2018 project in the future), which can benefit both PRA I&C model construction and verification. In the development also the following issues can be addressed from the integrated approach point of view; deterministic analysis of failure criteria and failure mode definitions.

Each task 1, 3 and 4 will be reported as separate reports or conference papers.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Jan-Erik Holmberg (Risk Pilot)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Markus Porthin, Kim Björkman (VTT)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>4.2 (Risk Pilot 1.5 pm, VTT 2.7 pm)</td>
</tr>
</tbody>
</table>
2.2 Analysis methods and tools (WP2)

While the purpose of WP1 is to set the common ground, WP2 goes deeper into assessment methods and tools for specific, technical, human and systemic, safety issues. What ties the methods to the common goal is that

- each method should apply the principles of structured argumentation on the basis of the gathered evidences
- each method should produce results that can be used as evidence in the higher-level safety demonstration

To provide transparent evidence for safety demonstrations, existing methods can be extended with suitable working methods and documentation practices. In addition, new assessment methods can be developed where needed. The main topics in WP2 are outlined below.

Topic: Integration of formal verification tools

Formal verification tools can be used as one method to provide evidence for safety demonstration, but their use so far has been quite isolated and laborious. This research theme aims for an integrated approach and toolset for safety demonstration in which the formal verification practices and processes are integrated to other related techniques and design principles. In practice this is realised as integration of V&V tools with vendor specific application development tools, integrating formal verification into I&C requirement specification processes, using simulators to validate the results of formal verification, and the co-use of deterministic and reliability based safety analysis methods to produce practical safety assessment approaches. This work results in more efficient and automatic verification practices, more efficient documentation of the results as well as novel approaches for safety assessment.

Topic: New assessment methods and tools

Novel methods and tools are developed in order to achieve a more extensive idea of plant safety, and to provide more evidence to use for safety demonstration. In 2017, the focus is System-Theoretic Process Analysis (STPA), and finalising of the Systems and safety engineering process assessment method. Examples of other potentially useful approaches include security assessment, assessment of DiD, multi-phased assessment of control room systems and I&C architectures and assessment of safety demonstrations delivered by the license applicant to the regulator.

WP2 will be coordinated by Antti Pakonen (VTT)

Partners and person months allocated to WP2 to be given in the table. The specific research tasks for 2017 are described in the sections below.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>2.5</td>
</tr>
<tr>
<td>Aalto</td>
<td>7.2</td>
</tr>
<tr>
<td>FiSMA</td>
<td>1.5</td>
</tr>
<tr>
<td>IntoWorks</td>
<td>1.5</td>
</tr>
</tbody>
</table>

2.2.1 Explicit state model checking (T2.1)

In the years 2015–2016 of SAUNA, the effort of T2.1 has been put on closed-loop modelling and formal verification by model checking, where, in addition to the controller model, a plant (or environment) model is used to specify feedbacks from the controlled process. By closing the control loop, unrealistic control scenarios are filtered out, which facilitates the later counterexample analysis process. Closed-loop verification allowed to meaningfully check a range of properties wider than the one possible for open-loop verification. However, closed-loop verification required significantly more time than open-loop one.

One of the main reasons for the increased verification time is the use of symbolic model checking. While this verification approach typically performs better than explicit-state model checking, it is unable to benefit from the state space reduction achieved by closing the loop. Thus, to make model checking more practical, T2.1 intends to compare the performance of explicit-state model checking with the symbolic one, which will require development of the corresponding model generation methods, patterns and tools.
Both SAUNA research so far and VTT’s practical customer projects since 2007 have largely been based on the NuSMV symbolic model checker. In particular, basic function blocks are modeled manually in NuSMV, after which function block diagrams are transformed to NuSMV by a tool (MODCHK) developed at VTT. The future of NuSMV, however, is to some extent uncertain. From a practical viewpoint, we should therefore investigate if and how other tools (such as SPIN, for example) can also be used in practical verification of NPP systems.

The work topics of this task for 2017 are aimed to reconsider the existing framework and thus include:

1. **Applying explicit-state model checking** for both open-loop and closed-loop model checking of nuclear automation systems, and comparing the results with the ones of symbolic model checking. The benefits and drawbacks of both explicit and symbolic model checking will be assessed. This topic, in particular, includes the considerations of feasibility of supporting explicit-state model checkers (such as SPIN and UPPAAL) in the employed toolset.

2. **Increasing the dependability of the basic function block library** by comparing them with the ones generated automatically from available specifications. Automatic generation of basic function blocks will be achieved by applying approaches of software model synthesis from behaviour examples and temporal (LTL) specification. The comparison of function blocks will be performed by behaviour equivalence checking, which can be handled by model checking. Such a comparison will allow identifying possible issues with the manually modelled basic function blocks.

3. In addition, any improvements of the NuSMV-based framework may be considered provided that they improve the extent to which the developed methods are practically applicable.

The research on these topics will be applied to the generic nuclear power plant model provided by Fortum Power and Heat Oy, which is implemented in the Apros simulation environment. Research into the generation of basic function blocks should preferably be based on actual specifications of, e.g., TELEPERM XS or Spinline systems, if possible.

During the work on the task, the Aalto side will collaborate with a research group of ITMO University (St. Petersburg, Russia), which also conducts research on closed-loop model checking of automation systems and has expertise in formal methods of verification and software model synthesis. For this reason, up to three research seminars in ITMO University are planned. These seminars will allow to get expert feedback on the ongoing developments and deliverables and new ideas which will facilitate the work on the task.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Valeriy Vyatkin (Aalto)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Igor Buzhinsky (Aalto), Antti Pakonen (VTT)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>6.5 (Aalto 5.0 pm, VTT 1.5 pm)</td>
</tr>
</tbody>
</table>

### 2.2.2 Requirement editing and refinement for formal verification (T2.2)

Earlier work done in the SAFIR2014 programme has shown that a collection of natural language requirement templates can be used to map the majority of typical (functional) I&C requirements into corresponding formal logic (Tommila & Pakonen 2014). However, the employed formal languages are often unfamiliar to most practitioners. Due to this reason, representing formal properties in user-friendly ways is one of the means of bringing formal methods closer to industrial practice. In the years 2015–2016 of SAUNA, the research on T2.2 concerned the examination of existing user-friendly property specification formalism and the investigation of temporal specification patterns common in the nuclear I&C systems domain. In 2017, research will target the following new topics:

1. **Concepts for requirement formalisation tools.** As shown in (Pakonen et al. 2016), no single solution (be it templates, patterns or a visual language) can alone be used to specify all the property types that occur when verifying (nuclear) I&C systems. While most properties can be expressed compactly using a small set of templates, a significant share (about 24%) deal with timing and sequencing issues that lead to complex formulas with several propositions (subformulas). Such formulas can rather conveniently be expressed using Property Specification Language (PSL). A tool concept combining the well-established idea of requirement templates with a new way of visually expressing PSL’s Sequential Extended Regular Expressions (SERE) style properties should therefore be explored.

2. **Structured demonstration of property satisfaction or violation.** In the most cases, when a temporal requirement is not satisfied, modern model checkers provide a counterexample which shows a concrete example of how the requirement is violated. VTT’s modelling tools (MODCHK) use 2D animation for visualising counterexamples, but from a complicated scenario, it may still be difficult to pinpoint when and
how exactly the property is breached. Thus, methods of advanced counterexample visualizations will be reviewed. One potential new feature is the “animation” of the property (requirement) formula during counterexample playback (visualisation of the truth value of each proposition (subformula) at each step).

The research on these topics will be aided by the generic nuclear power plant model provided by Fortum Power and Heat Oy. This model will be also used to illustrate the findings.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Valeriy Vyatkin (Aalto)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Igor Buzhinsky (Aalto), Antti Pakonen (VTT)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>3.2 (Aalto 2.2 pm, VTT 1 pm)</td>
</tr>
</tbody>
</table>

### 2.2.3 Application of System-Theoretic Process Analysis (STPA) in digital I&C (T2.3)

Systems-Theoretic Process Analysis (STPA) is a hazard analysis method that can be used both for existing systems in order to analyse their safety, as well as in the safety design of new systems, including potential for human errors. This application in new designs is called “safety-guided design process” (Leveson 2011).

During 2016, a Master’s Thesis was undertaken where the fitness of STPA safety guided design was evaluated on a nuclear plant refueling machine, or crane, from the point of view of nuclear safety. The case example was based on materials of an existing refueling machine, provided by TVO, which were used as background information. However, the design process made no assumptions about particular design choices, either existing or future ones, thus providing a high level analysis of the safety of such a hoist in the physical environment of the case, which is not changeable. The result of the task in 2016 were safety requirements and constraints on the high level functions of the refueling machine itself, provided in a systematic, traceable and justifiable way. However, during the Master’s Thesis work it was discovered that existing documentation and guidance for applying STPA in safety-guided design is lacking.

In 2017, this task disseminates and refines guidance on how to apply STPA for safety-guided design. As a method, STPA is found to be useful but due to its novel approach, deployment is challenging, particularly in design phase, due to lack of clear and unambiguous guidance. In the task, guidance is created based on lessons learned in 2016 as well as findings from the literature.

The result of this task will be a nuclear I&C designer’s guide to practical application of STPA in safety-guided design. The guide is intended to be easily approachable and to facilitate transfer of the knowledge gained in the course of the Master’s Thesis to the expert audience of Finnish nuclear I&C designers.

As a tertiary aim in 2017, we intend to investigate possibilities of international cooperation in the area of STPA application further after good initial contacts in 2016 with methodology experts and originators from the Massachusetts Institute of Technology.

Provided that early and positive results can be achieved, we hope to start a discussion for a more ambitious task of analysing an existing, complex system – including support functions by other systems – for the final year of 2018. This planning must be done in due time to assess needed resources, and to find and commit to a good case example.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Eero Uusitalo (IntoWorks)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Arttu Hirvonen, Mika Koivela</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>1.5</td>
</tr>
</tbody>
</table>
were presented. In addition, two research papers were published to discuss further development needs regarding the process assessment method. The Nuclear SPICE use cases were elaborated further in the research articles [Nevalainen et al. (2016); Varkoi, Nevalainen, Mäkinen (2016)].

Next, in 2017, main goals are 1) to extend the use of the Nuclear SPICE method; and 2) to study novel solutions for collection, management and reuse of the evidence data.

Firstly, new possible sources for assessable processes, e.g. the IAEA report Leadership and Management for Safety, and ISO/IEC 15026-4 Assurance in the life cycle, are analysed and further extensions to the assessment models are evaluated. A concise presentation of the applicability of process assessment is prepared to promote the use of Nuclear SPICE. This topic builds on the earlier work and will be reported in a research paper (D2.5.1) to be proposed for presentation at an appropriate conference, e.g. the ISOFIC 2017 International Symposium on Future I&C for Nuclear Power Plants, under the Safety Critical Software Development and Qualification topic.

Secondly, international research cooperation is continued to discover a novel approach to utilize the collected evidence data. The need for efficient evidence collection and management became evident when integration of process assessment and safety assurance was studied. An analysis of the assessment results provides the base data for a conceptual model. A research paper (D2.5.2) will define concepts for an evidence base and evaluate new database techniques that would enable a flexible and learning implementation. The paper is planned to be published at the EuroSPI 2017 conference, Standards and Assessment Models workshop.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Timo Varkoi (FiSMA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Risto Nevalainen (FiSMA)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>1.5</td>
</tr>
</tbody>
</table>

### 2.3 Safety demonstration practices (WP3)

The purpose of WP3 is to provide recommendations, insight and new viewpoints for

- planning, documenting and communicating the safety demonstration and its traceability links to system related artefacts
- enabling licensees to efficiently carry out the licensing process together with the regulator and suppliers.

In this work, WP3 uses and ties together the evidences provided by the assessment methods in WP2 and co-operates with the WP1 where the framework for the qualification activities is defined.

**Topic: Critical systems assurance**

Assuring, documenting and demonstrating the safety, security, reliability, or usability of critical systems requires well-defined and mature practices and processes together with computerised tool support. Traditional software and systems engineering techniques cannot provide the justified confidence needed. In addition, the large amount of loosely structured material, mainly in textual format, poses challenges for understanding, analysing and assessing the attributes of systems. The tasks under this topic during the whole 4-year project will concentrate on clarifying the current practises on systems assurance, main focus on safety. Different aspects to be covered will include the process of performing assurance along with licensing practises, structuring and presenting requirements and evidence, ways to argument and justify the solutions found and decisions made, and means of documenting it all to support all stakeholders immediately and after decades in various occasions. A model driven approach will be
utilised to capture the traceability links to requirements, design artefacts and test results. The focus will be kept in nuclear facilities, but supporting ideas and best practices will be acquired also from other critical areas of industry.

**Topic: Approaches supporting safety demonstration**

It has been a recognised fact that nuclear projects contain large amounts of documentation that is mainly based on narrative, references and links between different documents. More structured ways of presenting information and criteria of assessing the qualification material will be investigated under WP3.

**Topic: Practical examples for assurance**

After clarifying the current assurance practices within nuclear area and better understanding the weak points, potential improvements will be suggested and tried out in practical case studies. The case studies may be from real ongoing projects or past projects. The important thing is to find realistic examples where new concepts can be showcased.

Partners and person months allocated to WP3 to be given in the table.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months in 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>4.5</td>
</tr>
</tbody>
</table>

2.3.1 Overall plant automation safety demonstration (T3.1)

The overall I&C architecture of a nuclear power plant and its role in the Defence in Depth (DiD) concept are critical for plant safety and must, therefore, be decided early in the design and licensing process. For a convincing demonstration of safety on plant and system level, license applicants must combine analysis results provided by several engineering disciplines. Justification of the overall safety forms a hierarchical structure of claims, arguments and evidences concerning technical solutions, human organisations and development processes.

The objective of T3.1 is to define a top-level framework for safety demonstration and assessment of I&C architecture. The relevant (DiD) requirements are captured from regulations and standards (e.g. EPRI 2014) and organised as a set (pattern) of claims and assessment criteria following the principles of structured assurance cases (ISO 15026). The main purpose is to develop an approach to determine the quality of architectural specifications, with particular attention devoted to safety and DiD capabilities (from task T1.1), such that the architectural specifications and their assessment artefacts are traced to other artefacts from plant and safety engineering, I&C design and qualification.

The conformance assessment data model created in SAUNA’s Task T1.1 in 2016 (Alanen & Tommila 2016) is used as the basis for safety demonstration and assessment. This data model is discipline independent and assumes that in each discipline, the Systems Engineering core loop (see Figure 7) is executed. Now the plan is to verify the feasibility of the approach by applying it for the outcomes of the architecture definition process. This is done by demonstrating the model with an example (preferably a real industrial case), which ideally is shared with task T1.1 (MBSE methods for architecture level DiD analysis). This task (T3.1) is focused on qualification of DiD capabilities, while T1.1 provides the DiD verification and validation results used as evidence (i.e. T1.1 imitates the role of developers, while T3.1 imitates the role of assessors). In other words, the idea is to show that the DiD design and V&V artefacts (determination results) fit into the conformance data model. The example case is planned to be presented using modelling notation similar to UML object diagram or SysML requirements diagram.
The main deliverable (D3.1.1) of the task will discuss experiences of the safety demonstration approach in I&C architecture assessment and qualification, as well as tool support needed for managing all conformance assessment data. It is assumed that both practical insights and needs for improvement are derived from the example. Based on the feedback, change requests for updating the conformance assessment data model are issued.

To support the overall objectives related to safety justification in SAUNA, this task also includes cooperation with the Halden Reactor Project (HRP) to support their efforts on safety demonstration task in their current 3-year research programme. This work is part of the In-kind effort already agreed with (and funded by) HRP. The content of the work is to provide input and feedback to HRP while they are developing improved safety justification approaches and practical guidance on the topic. HRP is in co-operation with several international nuclear regulators, utilities, suppliers, research organizations and universities, and this activity provides a good possibility to get in the loop of international information exchange and bring the Finnish viewpoint in the discussions. In addition, participation in the meetings of the Halden Programme Group is included in this task.

This task has direct links to T1.1 (MBSE methods for architecture level DiD analysis), which provides the DiD architecture definition and analysis results. Related activities are planned in other tasks as well, e.g., in T1.3 (MODIG, DiD assessment) and in T2.5 (Process assessment for systems and safety engineering).

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Jarmo Alanen (VTT)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Participants</td>
<td>Teemu Tommila, Janne Valkonen, Joonas Linnosmaa, Hanna Koskinen (VTT)</td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>4.5</td>
</tr>
</tbody>
</table>
2.4 Project management (WP4)

2.4.1 Management and reporting (T4.1)

VTT will act as the project coordinator, managing communication with the project’s research group, other projects, the reference group guiding the project and the programme management, and is responsible for the reporting obligations set for the projects in the programme. Senior Scientist Antti Pakonen will act as the VTT project manager.

A co-operation agreement between SAUNA project group members was signed in 2015. The project managers of each member organisation (see Chapter 4.2) will share the responsibility for the actualisation of the research objectives.

T4.1 will also include the preparation of the project plan for 2018.

Project coordination will be carried out according to VTT practices. VTT’s operational system has been certified in accordance with the ISO 9001:2008 standard. The certificate was issued by DNV in 2006, and covers research, technology transfer and consultation services and the development of new technology at VTT.

<table>
<thead>
<tr>
<th>Task leader</th>
<th>Antti Pakonen (VTT)</th>
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</thead>
<tbody>
<tr>
<td>Participants</td>
<td></td>
</tr>
<tr>
<td>Person months in 2017</td>
<td>1.2</td>
</tr>
</tbody>
</table>

# Deliverables and milestones 2017

The deliverables and milestones planned for the project year 2017 are listed in the table below.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Milestone (T1.1):</strong> a) The development and results of the functional modelling based risk assessment method utilizing dynamic probabilities of external events will be ready. b) The interdisciplinary I&amp;C architecture assessment methodology will be in a good state.</td>
<td>1</td>
<td>30.9.2017</td>
<td></td>
</tr>
<tr>
<td>D1.1.1</td>
<td>Conference paper on functional modelling based risk assessment for early complex designs</td>
<td>1</td>
<td>31.3.2017</td>
</tr>
<tr>
<td>D1.1.2</td>
<td>Conference paper on an interdisciplinary modelling methodology for early I&amp;C architecture assessment.</td>
<td>3.3</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D1.3.1</td>
<td>A journal paper on “Digital I&amp;C design aiming at the creation of believable safety cases” (tentative working name)</td>
<td>1.5</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D1.3.2</td>
<td>Working report on the progress in the WGRISK digital I&amp;C benchmark study</td>
<td>1.2</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D1.3.3</td>
<td>Research report or a conference paper: Case study description – Coupling of PRA and model checking</td>
<td>1.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td><strong>D2.1.1</strong></td>
<td>A conference or journal paper on the comparison of explicit-state and symbolic model checking applied to nuclear automation systems</td>
<td>3.5</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D2.1.2</td>
<td>A conference paper on basic function block synthesis and the comparison of the synthesized blocks with manual implementations</td>
<td>2.5</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D2.1.3</td>
<td>A conference paper on the practical application of model checking in Finnish NPPs</td>
<td>0.5</td>
<td>30.6.2017</td>
</tr>
<tr>
<td>D2.2.1</td>
<td>A research report or a conference paper on the ways of making the demonstration of temporal specification satisfaction or violation more user-friendly</td>
<td>3.2</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D2.3.1</td>
<td>I&amp;C designer's guide for safety-guided design using STPA</td>
<td>1.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D2.5.1</td>
<td>A research paper: Process assessment for safety critical software.</td>
<td>0.5</td>
<td>30.11.2017</td>
</tr>
<tr>
<td>D2.5.2</td>
<td>A research paper: Evidence base concepts for combined safety assurance and process assessment data.</td>
<td>1</td>
<td>30.8.2017</td>
</tr>
<tr>
<td><strong>Milestone (T3.1):</strong> The Defence in Depth example case from T1.1 is integrated into a safety demonstration example that follows the conformance assessment data model.</td>
<td>1</td>
<td>31.10.2017</td>
<td></td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Example of model based safety demonstration/assessment of an I&amp;C DiD architecture design</td>
<td>3.5</td>
<td>31.12.2017</td>
</tr>
</tbody>
</table>

**Total pm:** 25.7
4. Project organisation

4.1 Project management

The organisation responsible for the coordination of the whole project will be VTT. Senior Scientist Antti Pakonen will act as the project manager at VTT.

4.2 Project consortium

The participating organisations (and the managers of the organisation-specific work) are:

1. VTT Technical Research Centre of Finland Ltd (Senior Scientist Antti Pakonen)
2. Aalto University (Doctoral Candidate Igor Buzhinsky)
3. Finnish Software Measurement Association FISMA (Senior Advisor Timo Varkoi)
4. Risk Pilot Ab (Office Manager Jan-Erik Holmberg)
5. IntoWorks (Partner Eero Uusitalo)

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Jarmo Alanen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1, T3.1</td>
<td>1.8</td>
</tr>
<tr>
<td>Antti Pakonen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T2.1, T2.2, T4.1</td>
<td>3.6</td>
</tr>
<tr>
<td>Nikolaos Papakonstantinou</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>1.8</td>
</tr>
<tr>
<td>Markus Porthin</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.3</td>
<td>1.2</td>
</tr>
<tr>
<td>Teemu Tommila</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1, T3.1</td>
<td>1.3</td>
</tr>
<tr>
<td>Janne Valkonen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1, T3.1</td>
<td>1.5</td>
</tr>
<tr>
<td>Kim Björkman</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T1.3</td>
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<tr>
<td>Hanna Koskinen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T3.1</td>
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<tr>
<td>Joonas Linnosmaa</td>
<td>Research Scientist</td>
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4.3 Related research projects

4.3.1 SAFIR2018 programme

In the SAFIR2018 research area “Plant safety and systems engineering”, there are four research project proposals that are linked by not only common themes, but also researchers working across projects, and joint work tasks.
**Probabilistic risk assessment method, development and applications** (PRAMEA, Ilkka Karanta / VTT) aims at developing methods for risk-informed decision making to support strategic and operative plant management. Although focus is on PRA methods, SAUNA and PRAMEA share the themes of safety assessment and Defence-in-Depth.

There will be concrete cooperation between SAUNA and PRAMEA, as the projects will have a joint work topic related to the co-use of PRA and model checking (T1.3). The MODIG plant model developed in SAUNA is highly important for PRAMEA, as well. WP2 of PRAMEA will also cover risk analysis of organisations and operation.

**Crafting Operational Resilience** (CORE, Jari Laarni / VTT) aims to promote nuclear safety and Defence-in-Depth by developing operational capabilities that support adaptability and flexibility. A key concept is that of resilience, the ability of a system to adjust to expected and unexpected changes and disturbances. CORE promotes resilience by developing tools and practices for operating personnel.

Cooperation between SAUNA and CORE has been established by key researchers working in both projects on tasks of joint interest.

**Management principles and safety culture in complex projects** (MAPS, Nadezhda Gotcheva / VTT) aims at identifying the safety principles of managing major projects in the nuclear industry, and clarifying the cultural phenomena and the variety in the involved actors on safety. Work on cultural dynamics complements related research on other (e.g., DiD and design process) analysis methods carried out in SAUNA.

CORE, PRAMEA and MAPS will all benefit from the work done on conceptual models (terminology) and the principles of Systems Engineering in WP1 of SAUNA.

![Figure 8. Common topics between SAFIR2018 projects in the “Plant safety” research area](image)

### 4.3.2 Finnish Research Programme on Nuclear Waste Management (KYT2018)

The goal of KYT2014 is to maintain national knowhow in nuclear waste management and to promote collaboration between authorities, nuclear industry, and scientists. In parallel to SAFIR2018, a new four-year programme period was started in 2015 with the acronym KYT2018. Within the new KYT programme, VTT and Aalto University have a project called **Systematic methodologies for safety case development** (TURMET). The project will use the principles of systems engineering and analysis for integrating long-term safety requirements to the safety case development in a transparent way. Moreover, the treatment of uncertainties in safety analysis is a scientific challenge considered in TURMET. With these goals, SAUNA and TURMET have common interests in the presentation and development of the safety case as part of the licensing process.
5. Risk management

5.1 Project's human resources

<table>
<thead>
<tr>
<th>Risk name</th>
<th>Bottlenecks of the resource base; availability &amp; necessary competencies</th>
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<tbody>
<tr>
<td>Description and impact</td>
<td>Key research objectives may not be reached if there are problems in human resources.</td>
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<tr>
<td>Mitigation</td>
<td>The project has a four year plan, and the SAFIR2018 reference group can modify the plans and budgeting in accordance to external pressures on resourcing. The project members have overlap of expertise, and loss of a single expert is generally not critical. VTT in particular has a large, multidisciplinary group of researchers involved in SAUNA, and funding can potentially be reallocated to tasks not short on resources, as long as key research objectives will not be compromised.</td>
</tr>
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<table>
<thead>
<tr>
<th>Risk name</th>
<th>Project group's ability to cooperate internally/ externally.</th>
</tr>
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<tbody>
<tr>
<td>Description and impact</td>
<td>The SAUNA project group consists of different organisations with different cultures and strategic objectives. This project is the first time that some of the organisational units are working together on a joint research project.</td>
</tr>
<tr>
<td>Mitigation</td>
<td>The project will specify clear objectives for each work task. To the extent that is possible, joint work is favoured over company-specific work tasks (although there will be VTT-specific tasks due to the volume and scope of VTT’s work). Responsibilities for work task leadership and work package coordination will be distributed between the organisations. A co-operation agreement was signed in 2015, specifying how the responsibility for the actualisation of the research objectives will be shared between the project group members.</td>
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</table>

5.2 Occupational safety, environmental and information security risks

<table>
<thead>
<tr>
<th>Risk name</th>
<th>Information security risks (computers/ PCs, tablets, smartphones, telecommunication, inadequate security culture etc.)</th>
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</thead>
<tbody>
<tr>
<td>Description and impact</td>
<td>Project data may include highly confidential information about the design and the operating processes of nuclear facilities. Unauthorised access to such data could have an impact on nuclear safety.</td>
</tr>
<tr>
<td>Mitigation</td>
<td>At VTT, data security forms part of the overall security of the VTT Group, which is the responsibility of the President and CEO. Every VTT employee is responsible in his/her work and his/her activities for data protection, regardless of the content or form of the data. Risk Pilot is used to working with classified material, and fulfils the requirements of the Swedish utilities on data security. IntoWorks has competence and means for working with classified materials. Aalto implements full scale computer security measures aiming at protection of the sensitive and confidential information. FiSMA personnel are certified process assessors and accustomed to work under bilateral confidentiality agreements.</td>
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### 5.3 Risk checklist

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<th>Relevant?</th>
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<td><strong>Project objectives, definitions, tasks, maturity of the relevant technology</strong></td>
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<tr>
<td>Technology, that is very new, fluid or difficult to obtain is developed or used in the project</td>
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<td>Critical tasks or other elements that have significant impact on the success of the project can be identified.</td>
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<tr>
<td>Information/ data that can substantially affect project’s progress or objectives (e.g. when becoming more accurate) is acquired/ collected for the project</td>
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<tr>
<td>External mandatory rules and regulations (Customer, legislation, other?)</td>
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<tr>
<td>IPR rights/ restrictions</td>
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<td><strong>Project’s human resources</strong></td>
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<td>Bottlenecks of the resource base; availability &amp; necessary competencies, possible substitute arrangements</td>
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<td>Management of time usage.</td>
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<tr>
<td>Understanding and acceptance of the project objectives and research methodology (commitment of the project group)</td>
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<tr>
<td>Project group’s ability to cooperate internally/ externally.</td>
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<tr>
<td><strong>Timetable or cost pressures and financing</strong></td>
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<tr>
<td>The project has challenging timetable and cost targets with no flexibility (project group’s influence potential vs. external pressures and boundary conditions).</td>
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<tr>
<td>Are there connected sub-projects (or similar) that are potentially on a critical path.</td>
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<tr>
<td>Possibilities/ limitations to align timing (etc.) with other unfinished or planned projects.</td>
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<tr>
<td>Vulnerability to impacts/ changes in external conditions.</td>
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<tr>
<td>Certainty and timing of the project financing, critical time limits/ deadlines.</td>
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<tr>
<td>Importance of the project for the (main) financing organization.</td>
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<tr>
<td>Realism and accuracy of the project budget.</td>
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<tr>
<td><strong>Project’s external stakeholders / cooperation with subcontractors</strong></td>
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<td>Subcontractor’s project management skills, engagement with processes of the ordering party.</td>
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<td>Technological capabilities of the subcontractor.</td>
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<td>Match of the organization cultures.</td>
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<td>Subcontractor’s ability to meet security/ safety standards.</td>
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<td>Financial position of the subcontractor.</td>
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<td><strong>Equipment, premises and infrastructure (project/ research environment)</strong></td>
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<td>Priority order in case there are other projects that compete of use of the same research environment.</td>
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<td>Limited or no flexibility re. critical infrastructure.</td>
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<td>Dependence on completion of some other project, piling up of the work, ‘domino effect’</td>
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<td>Bottlenecks of the investment decisions.</td>
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<td>Possible equipment failures that may lead to interruptions or delays in the research work, interruptions in water/ electricity distribution, data protection issues (back-up copying etc.)</td>
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<td>Negative environmental impacts (radiation, chemicals, excessive noise…)</td>
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References


MODIG-PLANS workshop, 28-29.9.2015, Espoo, Finland.


### Work packages and Tasks

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Comments:

Travel expenses T1.1 Conference travel
T1.3 Conference travel, OECD WGRISK meeting
T2.1 10th ANS NPIC&HMIT, San Francisco, US, June 11-15
T2.2 Conference travel
T3.1 10th ANS NPIC&HMIT, San Francisco, US, June 11-15
HPG meetings in Germany and Norway, Apr&Sep; IAEA TWG-NPPCI meeting in Vienna, May

Other costs T4.1 Catering for the reference group meetings etc. arranged by SAUNA
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Comments:

- Travel expenses
  - T2.1 INDIN 2017, ETFA 2017 conferences, seminars in ITMO University
  - T2.2 INDIN 2017, ETFA 2017 conferences
## FiSMA

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### Comments:

- Travel expenses T2.5: Two international conference attendances, project meeting travel costs.
- Other costs T2.5
- Indirect personnel cost rate: 50%
- Over head cost rate: 30%
## Risk Pilot

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**Comments:**

- Travel expenses T1.3: Project meetings in Sweden

Indirect personnel cost rate: 50%

Over head cost rate: 30%
## IntoWorks

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**Comments:**
- Travel expenses T2.3: TR meetings (from Oulu etc), expert meetings, possible conference or workshop (in part).
- Materials and supplies T2.3: ISO / IEC / IEEE Standards, etc
- Indirect personnel cost rate: 50%
- Over head cost rate: 30%
Steering Group SG2 – Reactor safety:
SAFIR2018 Project plan

CASA

Comprehensive Analysis of Severe Accidents

Anna Nieminen, Tuomo Sevón, Jukka Rossi, Mikko Ilvonen, Veikko Taivassalo, Magnus Strandberg

VTT Technical Research Centre of Finland Ltd
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1. Research theme and motivation

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.

1.1 Background and state-of-the-art

Despite of nearly 40 years of research in the area of severe accidents there are still plenty uncertain issues. This is partly because large scale experiments with real materials are extremely difficult, if not impossible, to execute. In addition, many different phases in the progress of a severe accident are so tightly linked together that it is not possible to obtain totally reliable information based only on separate effect tests. The majority of the research in this area is done by simulations. Thus, it is extremely important that the simulation codes and methods are validated for their intended purposes and that the users of the codes have expertise to apply the codes to reliable assessment of nuclear safety.

Lessons learnt from Fukushima accident

Fukushima accident provides a unique opportunity for gaining more information on the progress of severe accidents and their prevention and mitigation. Detailed analyses of the Fukushima accident have only started (Sevón, 2015a; Sevón 2015b), and the knowledge and the gained experience that can be utilized to improve NPP safety should be made available to the Finnish authorities, power utilities and the research community. It is important to stay in the international community leading the research and absorb the relevant knowledge of the accident.

The Fukushima accident has also highlighted the vulnerability of nuclear fuels stored in spent fuel pools (SFP) due to a potential loss of sufficient cooling. Depending on the codes and modelling hypothesis, the conclusion on SFP coolability can be very different (Fleurot et al., 2014). The reliability of the results obtained with integral severe accident codes is questionable regarding in particular the following phenomena: natural convection and boiling, conditions of air ingress, cladding behaviour in the presence of air and coolability of dry fuel.

After Fukushima some efforts have been directed also to the development of fuels with enhanced accident tolerance (Zinkle et al., 2014). This might mean slower reaction kinetics with steam, slower hydrogen generation rate or enhanced retention of fission products, i.e. improved fuel pellet and cladding properties. Possibilities to benefit from Accident Tolerant Fuels (ATFs) in severe accident management should be evaluated concerning the Finnish NPPs.

Core melt management

The main focus in severe accident management is in the core melt cooling. Phenomena related to cooling of intact core are well-known, but the efficiency of degraded core cooling is uncertain (Steinbrück et al., 2010). This is partly because also the behaviour of damaged core is uncertain.

Claddings will lose most of their mechanical strength after a significant oxidation. Fuel rods are likely to collapse and form a particulate debris bed in particular at the time of reflooding. In the next phase molten pools may form in the core. If reflooding is unsuccessful, the pool grows axially and radially and then relocates. Predictions relating to mass, composition and temperature of the material relocated to the lower head, as well as relocation times, are critical in evaluating the further accident scenario. Models have been developed and validated in most codes (Bandini et al., 2010) but the simulation results are not yet satisfactory relative to the experimental data available.

If accident proceeds to late-phase, the management strategy depends on the reactor concept. The core melt can be stabilized either in- or ex-vessel. The feasibility of In-Vessel Melt Retention (IVMR) has been demonstrated before for certain reactor types by proving that when the melt is located in the Reactor Pressure Vessel (RPV) lower head, the heat flux from the vessel wall does not exceed the critical heat flux. However, recent research on corium behaviour (Bechta et al., 2008; Seiler et al., 2007) has showed that it is important to review, for example, how the
heat flux profile behaves in transient conditions i.e. in a situation where the metal phase is initially heavier and eventually lighter than the oxide phase.

The integrity of the RPV is not affected only by the thermal loading but mechanical loading as well. The failure mode of the vessel is related to vessel material properties, thermo-mechanical loading and the load rate caused by the discharging core melt (Koundy, et al., 2008). The integrity of the vessel can be assessed with the Finite Element Method (FEM) codes more reliably than with integral codes. Thermo-mechanical simulations provide details of the vessel rupture mode (Nicolas, et al., 2003). Estimating the timing and the place of vessel rupture define eventually the containment loadings caused by the discharging core melt.

If the RPV integrity is lost, the long-term cooling of the melt must be ensured ex-vessel. Flooding the containment lower drywell has been adopted as the severe accident management strategy at the Nordic BWRs. The removal of the decay heat from the ex-vessel corium in the water pool has to be ensured to avoid possible basemat melt through. The melt will fragment and solidify when discharged to water forming eventually a porous debris bed to the bottom of the drywell. The coolability of the debris bed depends on several factors which cannot be exactly predicted due to the randomness of the debris bed formation process and the differences in accident scenarios (Kudinov et al., 2014). The coolability of ex-vessel core debris has been studied at VTT with the COOLOCE test facility which is the only existing test facility in which the debris bed shape and its variations are taken into account (Takasuo et al., 2012). Traditionally the coolability has been evaluated based on the dryout heat flux, but recently the focus has been on analysing post-dryout temperature behaviour of the bed (Yakush and Kudinov, 2014).

Generation III reactor designs rely typically on ex-vessel melt retention. This is done by implementing an additional safety barrier between the RPV and containment, i.e. a core catcher. Depending on the design, their operation has not yet been verified in all possible circumstances among the scientific community. The only published core catcher experiment with real reactor materials is the WCB-1 test in Argonne National Laboratory (Farmer et al., 2009).

**Ensuring the containment integrity**

Ensuring the integrity of the containment during a hypothetical severe accident is extremely important since the containment is the last safety barrier preventing radioactive release to the environment. In addition to core melt also highly energetic events steam and hydrogen explosions may threaten the containment integrity. In the case core melt is discharged from the RPV to a water pool, melt will fragment to water causing a rapid transfer of thermal energy. This leads to a major pressure increase and in certain conditions this may lead to a steam explosion that could lead to early containment failure. Based on current research, it cannot be confirmed in which conditions the explosion is triggered (Leskovar & Uršič, 2008).

In Finland, hydrogen explosions are excluded either by inerting the containment with nitrogen or by hydrogen combustion in a controlled way. Despite hydrogen recombiners, it might be possible that into some part of the containment is formed a flammable gas mixture. A local ignition might first result in a slowly propagating flame which can be quenched or accelerated depending on the gas composition, turbulence conditions and the direction of flame propagation. The containment sprays that decrease the pressure have an effect on turbulence, and thus on the containment loadings caused by hydrogen combustion.

The basic phenomenology of gas distribution and combustion is well understood. However, there is a lack of knowledge of gas distribution mechanisms for example during simultaneous action of multiple mitigation systems and of gas combustion in complex multi-department geometries. Despite continuous validation and development of analysis tools, the performance of the CFD and lumped parameter codes has been found limited (e.g., Baraldi et al., 2010; OECD/NEA 2012) especially for a system for which no experimental data is available. That is especially true for actual containments, as the current knowledge is based on experiments in facilities that are some orders of magnitude smaller.

**Fission product behaviour in the containment**

In addition to ensuring the containment integrity also fission product behaviour inside the containment has an important effect on the potential source term to the environment. This is important also in accident scenarios with controlled containment pressure relief, since all fission products cannot be trapped with Filtered Containment Venting (FCV) systems. Venting lines may include filters that are able to retain almost all aerosol particles but only a small part of gaseous iodine and negligible part of organic iodides that are produced mainly by interaction of iodine with paints. Iodine chemistry in the containment is considered extremely complex (Clément et al., 2007; U.S.NRC, 2013).

An important mechanism for trapping fission products is pool scrubbing that takes place in the BWR suppression pool, in the PWR pressurizer relief tank, in the case of steam generator tube rupture, in FCV systems having wet scrubbers and in a containment water pool covering core melt. The present pool scrubbing models for retention of
aerosols and gas phase radionuclides (I\textsubscript{2}, organic iodides, HOI, HI) have been developed based on experiments done in the 1980s and 1990s. These models are rather simple and they should be evaluated again based on current improved knowledge (Herranz et al., 2014). There is also lack of information of pool scrubbing phenomenon at elevated pool temperatures.

Fission product behaviour also affects dose rates in the containment that on the other hand have effect on iodine chemistry. Doses might affect the operation of instrumentation and automation systems and leak-tightness of containment penetration seal materials. To be able to assess the operation of these systems and structures in all circumstances, different ways for dose rate definition should be evaluated and compared. (Kalugin, 2008) At the moment at VTT dose rates can be produced (1) directly with integral code ASTEC; (2) using “Infinite Cloud” method for air volume and the point-kernel results assuming infinite water reflection for water pool; and (3) with MCNP code. Both latter options however need the source term pre-defined and re-processed with for example ORIGEN2 code. In addition to previous methods, the objective of the SAFIR2018 project RADICAL is to develop Serpent into a practical tool for radiation dose rate calculations in 2017.

**Environmental consequences**

If the containment integrity is lost, it is necessary to assess the transport of radioactive release to evaluate environmental consequences. There are number of atmospheric dispersion models applied to calculate for example dispersion and doses. Simple Gaussian dispersion models are widely used in probabilistic consequence assessments. The more advanced models can use weather data from large areas and even numerical weather forecasts to estimate long-range spreading of particles and gases in the atmosphere. (Rossi and Ilvonen, 2015)

In the case of radioactive release from a NPP there may be need to make protective actions in the environment. Protective measures depend on the release magnitude and are primarily needed in the vicinity of the power plant. However, as Chernobyl and especially Fukushima accident proved (Shorijo, 2014; EURATOM/2013/59), protective measures are needed also at longer distances. Deterministic effects are not expected at longer distances, but protective measures there could reduce the risk of stochastic effects (WHO, 2012).

Now IAEA recommends two additional planning distances beyond the former emergency planning zone. In Finland the Radiation and Nuclear Safety Authority (STUK) is responsible for preparation of instructions and orders to protect population, agriculture and other important activities in the society. Regional rescue authorities are responsible for implementation of the protective measures in radiation accidents. Therefore, current studies of the expected doses beyond 20 km are needed. (IAEA, 2014).

1.2 **Objectives and expected results**

The objective of this project is to develop safety analysis methods which benefit the safe and sustainable use of nuclear energy in Finland. The capability of simulation tools in use, including integral codes and several specialised programmes to model phenomena related to severe accidents will be assessed. If needed, the codes and methods will be further improved in collaboration with colleagues around the world. Reinforcing international networks will bring the most recent relevant knowledge to the use of Finnish nuclear community. The objective is not only to follow the international actions but to adopt the latest information to Finnish context.

The main outcomes of the project are (1) comprehensively validated simulations tools available for the assessment of severe accident scenarios for the needs of the Finnish NPPs; (2) trained experts who can use these tools and have in-depth understanding on the complex physical phenomena; and (3) significant reduction of uncertainties in the key processes that determine the consequences of a severe accident.

The results of the project will be published in the form of articles in scientific journals and conferences and as theses and dissertations of undergraduate and doctoral students. Thus, the high scientific quality of the results is ensured. At the end of this project, Finland will be able to more reliably analyse severe accident scenarios in our current and future NPPs.

1.3 **Exploitation of the results**

Knowledge of the different phenomena and awareness of the remaining uncertainties, as well as their management strategies, forms the basis for decision making concerning severe accident management. When needed, the project aims to build a comprehensive review of the status of the research and always answer the key questions especially from the Finnish point of view. Performed safety analyses, gained expertise and verified simulation methods are useful for Finnish authorities and power utilities immediately after the results are presented. The information produced in this project will help to assess the feasibility or adequacy of severe accident management procedures or safety systems.
Most of the phenomena and the technical solutions examined in the project are applicable to all NPPs operating in Finland. There are also issues more oriented to certain plant types: e.g. Olkiluoto BWRs benefit from ex-vessel debris coolability and steam explosion analyses whereas the core catcher concept is interesting e.g. for EPR and AES-2006. Also, the international connections of the project, which largely consist of the networks established in the previous SAFIR programmes, promote the international recognition of the Finnish research. By active information exchange with the international partners, the research done in the project can also benefit the international severe accident research.

1.4 Appropriateness of the project to SAFIR2018 programme

In the frame of this project several codes and current analysis methods will be validated. New experts are educated to master the use of these codes and phenomena related to severe accidents. The objective is to produce experts with a wide understanding of nuclear safety. This is partially done by bringing together various experts whose special know-hows are not necessarily directly related to severe accident applications. The personnel of this project will also be active in the international community of severe accident research by participating to several programmes, which include also experimental activities.

The topics of this project cover most of the subjects pointed out in SAFIR2018 framework plan and its update assessed as being of research priorities related to the mitigation of severe accidents: the project not only enables the use of integral codes but also trains new code users, core melt coolability will be assessed in possible locations, possibilities to analyse steam explosions will be developed, pool scrubbing related phenomena will be analysed diversely, ways to evaluate dose rates in different containment volumes will be assessed and also methods to analyse the transport of radioactive releases in the atmosphere will be improved.

1.5 Education of experts

One of the main goals of the project is to maintain and develop expertise related to severe accidents on a wide range. This project improves skills in the use of integral and several specialized codes and trains new code users and young researchers. The project personnel also include experienced researchers who work in co-operation with the younger generation which benefits the knowledge transfer.

The objectives for education of experts of the project include at least one PhD and one master’s thesis.

The project manager is a young researcher for who the project offers a good opportunity to develop her skills in leadership, project management and professional networking.
2. Work plan

2.1 Progress of severe accidents (WP1)

Tasks in WP1 will aim to overall understanding of the progress and mitigation measures of severe accidents. The focus will be in phenomena highlighted by the Fukushima accident.

MELCOR models of Fukushima unit 1, 2, and 3 accidents were developed in the SAFIR2014 programme. In SAFIR2018 these models will be improved, using new information that is being released by TEPCO. Objectives of the work are: (1) improving expertise in severe accident modelling, using data from a real full-scale reactor accident; (2) gaining a better understanding of the events in the Fukushima reactors; and (3) getting insights into the capabilities and weaknesses of the MELCOR code in simulating severe accidents. In 2015 a paper of the Unit 1 model was published in Annals of Nuclear Energy, and the Unit 2 model was updated. In 2016 the Unit 3 model was updated. The work has highlighted the importance of stratification of the suppression pool on the containment pressure and the challenges related to modelling of the stratification. Participation fee of the OECD/NEA BSAF-2 (Benchmark Study of the Accident at Fukushima) project was paid in total already in 2015.

The CSARP agreement provides license to use the MELCOR code for all Finnish nuclear energy organizations.

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2.1.1 Fukushima (T1.1)

VTT and STUK are members of the OECD/NEA BSAF-2 project, which will continue until 2018. Participation in BSAF-2 provides us access to more detailed plant data and gives the chance to cooperate with other organisations that are modelling the accident.

In 2017 the MELCOR model of one of the units will be updated, using newly available plant data. A poster about the Fukushima calculations will be prepared for the ERMSAR conference in Warsaw in May 2017. The BSAF-2 project meetings will be participated.

2.1.2 CSARP (T1.2)

Via U.S.NRC CSARP (Cooperative Severe Accident Research Program) the latest versions of the integral severe accident analysis program MELCOR are got into use for the Finnish nuclear energy organizations. Finland’s MELCOR license fee, 35 kUSD, is paid as part of this task. Furthermore, the annual CSARP/MCAP (MELCOR Code Assessment Program) meeting will be attended.

2.2 Core melt management (WP2)

In WP2 the core melt coolability will be assessed in ex-vessel configurations. The remaining uncertainties in coolability will be investigated in co-operation with European colleagues. The aim is to prove the feasibility of melt stabilization strategies in plant designs relevant in Finland.

Debris bed coolability has been several years the most important topic for co-operation between VTT and KTH in the frame of NKS projects DECOSE (DEbris COolability and Steam Explosion, 2012-2015) and SPARC (Scenarios and Phenomena Affecting Risk of Containment failure and release characteristics, 2016-2019). The collaboration offers a valuable view into the extensive severe accident research of KTH, which includes e.g. experiments on debris bed spreading and the integration of probabilistic and deterministic methods to estimate the risk of containment failure. One of the objectives in the new SPARC project is to tighten the co-operation between researchers working on deterministic analyses and PSA.

In 2015, VTT’s analyses on the debris bed shape were concluded by writing a scientific paper synthesising the results of the COOLOCE experiments and by preparing a doctoral dissertation. In the thesis was stated that the coolability of the debris bed depends on both the flooding mode and the height of the bed: multi-dimensional flooding increases the dryout heat flux and enhances coolability but in heap-like beds larger steam fluxes take place at the top counteracting the improved cooling effect of the multi-dimensional flooding. As a result, the maximum height of
a heap-like bed can only be about 1.5 times the height of a top-flooded cylindrical bed in order to preserve the direct benefit from the multi-dimensional flooding.

Based on suggestions made in the doctoral thesis, in 2016 the coolability of a multi-dimensionally flooded conical debris bed was estimated less conservatively establishing a temperature-based dryout criterion. First the capabilities of models in use to predict the post-dryout temperature were evaluated and no notable differences was observed in models for the effective thermal conductivity or convection. The temperature was found to stabilize even when half the bed height was dried out. This means a relative power increase of more than 50%. VTT’s MEWA results on post-dryout temperature were compared to KTH’s DECOSIM results. A good agreement was achieved while the temperatures continued to increase, but the stabilized temperatures differed notably. The cause for the differences should be examined.

In 2015 the performance of core catcher concept was assessed analysing WCB-1 experiment that is the only published core catcher experiment using real reactor materials. Heat fluxes and heat transfer coefficients were calculated and compared with available correlations. It was found out that the correlations overestimate heat transfer coefficients and that heat flux is more biased upwards than expected from correlations.

Work on core catcher analyses was continued in 2016 by testing MELCOR’s new melt pool coolability model with the water ingress mechanism by calculating the SSWICS and CCI experiments made in the OECD MCCI project. The new water ingestion model had a surprisingly small effect on the results of the SSWICS experiments that were made without gas sparging through the melt pool from decomposing concrete. For CCI experiments, where gases were released, the new model gave satisfactory results. The results were published in the Nuclear Technology journal.

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### 2.2.1 Particle debris coolability (T2.1)

Work on evaluating the coolability of a multi-dimensionally flooded conical debris bed less conservatively estimating a temperature-based dryout criterion is continued. The focus will be on analysing the temperature when it begins to stabilize since notable variation in results between KTH and VTT was observed. It is evaluated, if more information on the temperature behaviour could be achieved with CFD analysis. This task is a part of the NKS-SPARC project.

### 2.3 Containment Phenomena (WP3)

In WP3 the biggest threats for the containment integrity are analysed. In addition to core melt these include highly energetic events steam and hydrogen explosions. These phenomena have been also part of the collaboration between KTH and VTT partly already in the frame of NKS-DECOSE and more strongly in NKS-SPARC project. Typically, steam and hydrogen explosions can be analysed only on a very scarce level if at all with integral codes. To achieve reliable results specific know-how related not only to the phenomena but also to specialised codes is needed. However, integral codes are needed to examine the possible accident scenarios to evaluate the possibility of these phenomena to occur.

In 2015 steam explosion loads in Nordic BWR geometry were assessed and sensitivity of the results to key input parameters was examined using MC3D code. First, the effect of triggering time was analysed. The results showed that as long as the mixture is triggerable the strength of the resulting explosion does not change notably. Sensitivity analysis results showed that the melt drop size that is dependent on the physical properties of the melt had the strongest effect on the explosion strength. Surprisingly, the melt temperature did not affect the explosion strength as long as the temperature was high enough to cause an explosion.

In 2016 the focus was on hydrogen explosions. Accident scenarios that may lead to hydrogen explosions in the Nordic BWR containment and reactor hall were examined. Hydrogen explosions are possible in the containment only if the inertion is lost. 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 effect on the timing of the radioactive release and to 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 high level increasing the leak.
Remaining open topics related to hydrogen and fission product issues in the containment are being further investigated in the OECD/NEA THAI-3 (Thermal-hydraulics, Hydrogen, Aerosols and Iodine) project. The focus is in four topics that are experimentally investigated: (1) PAR (Passive Autocatalytic Recombiner) performance under counter-current flow condition; (2) hydrogen combustion and flame propagation in the two-compartment system; (3) fission product re-entrainment from a water pool at elevated temperature; and (4) re-suspension of fission product deposits upon impact of a high-energetic event, e.g. hydrogen deflagration. The experimental data will support validation and further development of lumped parameter and CFD-based models in the fields of thermal hydraulics, hydrogen deflagration, PAR modelling, aerosol behaviour and iodine chemistry. Additionally, the coupled-effects nature of the experiments are valuable in improving the prediction capabilities of the severe accident analysis tools. There are also analytical activities related to all experiments in the THAI-3 project that should be participated to get added value from the project.

In the SAFIR2018 CATFIS project pool scrubbing experiments are planned for 2017 to have more complete understanding of the related phenomena. The experiments will be performed both with CsI aerosol particles and with gaseous organic iodine CH₃I. The pool temperature varies from 20 °C to 100 °C and also the effect of pool pH is tested. Simulating the experiments would help to understand more profoundly the observed phenomena. In addition, the planned test matrix offers excellent validation data for the integral codes what comes not only to pool scrubbing but also iodine liquid phase chemistry.

Well-founded dose estimates are needed when licensing the operation of instrumentation and automation systems and containment penetration seal materials under severe accident conditions. Ways to evaluate dose rates in different containment volumes have been evaluated and compared. In 2015 ASTEC input deck was created to produce dose rates in the Nordic BWR containment. The maximum dose rates were recorded in the drywell and the dose rates on walls were almost three times higher than in the gas phase. It was concluded that defining dose rates on a too simple method may give unrealistic results. For example, using too large time steps may result missing the dose peaks and assuming even spreading to all containment volumes would also give misleading results. In 2016 the dose rates produced with ASTEC were compared to dose rates defined using “Infinite Cloud” and point-kernel results methods. Comparing these results to a Monte Carlo based method (MCNP or Serpent) will be continued in 2018.

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2.3.1 Steam explosions (T3.1)

The effect of RPV breaking mode on dynamic pressure loads on cavity wall induced by steam explosions will be analysed with MC3D. The focus will be on evaluating the effect of break location and size. The cases will be carefully assessed to correspond to realistic conditions. Previously studying the effect of break location was unsuccessful: despite high explosivity value the mixture was not able to be triggered. The reason for this unphysical result should be studied and the analysis completed. This task is a part of the NKS-SPARC project.

2.3.2 Pool scrubbing (T3.2)

Some of the pool scrubbing experiments made in the SAFIR2018 CATFIS project will be simulated with ASTEC and MELCOR. The focus is in evaluating the code capabilities to simulate not only the decontamination factor but also iodine liquid phase chemistry. Especially on the effect of pH there is a lack of validation data.

As a part of this task is prepared a paper on the effect of pH control on iodine behaviour based on previous work for the ERMSAR conference in Warsaw in May 2017.

2.3.3 OECD/NEA THAI-3 (T3.3)

Finland is participating to OECD/NEA THAI-3 project and the participation fee is paid from the CASA project budget. In addition to following the experimental activities, analytical activities are planned to participate to get added value. In 2017 will be organised an open simulation exercise on fission product re-entrainment from a water pool at elevated temperature. Analysing these experiments that represent long-term severe accident conditions is planned with ASTEC to complement the understanding on code capabilities achieved in simulating the VTT’s pool scrubbing experiments.
2.3.4 Hydrogen combustion benchmark (T3.4)

IRSN is organising a hydrogen combustion benchmark in the frame of the MITHYGENE project. Nine organisations from seven countries will be involved and ETSON has agreed to support the benchmark with compensating the travel costs to the meetings. Three hydrogen combustion experiments will be simulated as a double-blind validation study on flame propagation in the new ENACCEF2 facility. The new facility has more simple geometry than the THAI-facility but it includes obstacles that accelerate the flame propagation as real containment rooms. Therefore, the experiments provide important information on realistic cases.

2.4 Environmental consequences (WP4)

There are two primary models at VTT available to assess the environmental consequences for emergency preparedness purposes and level 3 PSA. ARANO is a straight line Gaussian type dispersion model for probabilistic consequence assessments where weather remains the same until the plume exits the computation area. VALMA is a dispersion and dose assessment code purposed to serve as an emergency preparedness tool that is able to use many kinds of weather data.

Previously the capabilities of these codes to estimate the needed countermeasure actions beyond the emergency planning zone (20 km) have been assessed. In 2015 the VALMA model was modified to calculate results with a probabilistic approach and dose results were calculated up to 300 km for one-year weather data with three different source terms. The results indicated that if the release magnitude exceeds significantly the criterion value for a severe accident, countermeasures may be needed up to 200 km in order to reduce stochastic health effects as IAEA recommends. In 2016 VALMA was augmented with ingestion dose pathways and was shown to be able to produce ccdf distributions of doses like ARANO does. Taking into account the ingestion doses affects also the area where countermeasures are needed, typically extending it.

In recent years after Fukushima, there has been some renewed interest in atmospheric dispersion and dose assessment. This interest even extends to full-scope level 3 PSA, e.g. in Japan under new stricter regulations. The competence to perform level 3 PSA with the codes ARANO and VALMA should be further improved.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.3</td>
</tr>
</tbody>
</table>

2.4.1 Emergency preparedness in cases of radioactive release (T4.1)

VALMA will be augmented with the calculation of acute and late health effects of radiation doses to increase also its competence to assist in radiation protection and in level 3 PSA. The probability distributions of radiation-induced health effects at various distances and over various time periods will be produced. The results will be compared to some corresponding ARANO analyses. VALMA results are expected to be more reliable since the code utilizes realistic weather data.

2.5 Project administration (WP5)

This work package contains the administrative duties of the project and other project management tasks which will be performed by the project manager.

<table>
<thead>
<tr>
<th>Partners in WP5</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
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</tr>
</tbody>
</table>

2.5.1 Project management (T5.1)

The project manager is responsible for supervising the progress of the different tasks and the realisation of deliverables. She is in charge of the communication with the reference group and project management as well as of the communication within the project team.
The project manager will take care of the administrative reporting including e.g. the progress and summary reports requested by the SAFIR2018 management. The manager will attend to the meetings where the project progress is evaluated and prepare presentation materials for the steering and reference group meetings. She is also in charge of writing the yearly project plans and updating the plan and budget.
## 3. Deliverables and milestones 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
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<tbody>
<tr>
<td>D1.1.1</td>
<td>Research report of improved Fukushima calculations</td>
<td>1.6</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D1.1.2</td>
<td>ERMSAR conference poster: Modeling of the Fukushima accident with MELCOR</td>
<td>0.4</td>
<td>31.5.2017</td>
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<tr>
<td>D1.2.1</td>
<td>Latest versions of MELCOR available for Finnish nuclear energy organizations</td>
<td>0.1</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D1.2.2</td>
<td>Presentation in CSARP/MCAP meeting</td>
<td>0.3</td>
<td>31.9.2017</td>
</tr>
<tr>
<td>D1.2.3</td>
<td>Travel report from the CSARP/MCAP meeting, summarizing the most interesting presentations</td>
<td>0.2</td>
<td>31.10.2017</td>
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<tr>
<td>D2.1.1</td>
<td>Research report on the post-dryout behaviour of multi-dimensionally flooded debris beds</td>
<td>1.4</td>
<td>31.9.2017</td>
</tr>
<tr>
<td>D2.1.2</td>
<td>Distribution of NKS and KTH reports</td>
<td>0.1</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Research report of the effect of vessel breaking mode on dynamic pressure loads induced by steam explosions</td>
<td>1.5</td>
<td>31.9.2017</td>
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<tr>
<td>D3.2.1</td>
<td>Research report or scientific publication of simulating the pool scrubbing experiments with ASTEC and MELCOR</td>
<td>3.0</td>
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<td>D3.2.2</td>
<td>ERMSAR conference paper: The effect of post-accident pH control on iodine behaviour in the containment</td>
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</tr>
<tr>
<td>D3.3.1</td>
<td>Distribution of THAI-3 reports</td>
<td>0.1</td>
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<td>D3.3.2</td>
<td>Presentation of the fission product re-entrainment benchmark results in the THAI-3 PRG4 meeting</td>
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<tr>
<td>D3.4.1</td>
<td>Contribution to the common MITHYGENE ETSON benchmark deliverable</td>
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<tr>
<td>N/A</td>
<td>Milestone: ARANO analyses finished</td>
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<td>30.9.2017</td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: Health effect models added to VALMA</td>
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<tr>
<td>D4.1.1</td>
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</table>

**Total pm** 13.2
## 4. Project organisation

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2016)</th>
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<tbody>
<tr>
<td>Ilvonen Mikko</td>
<td>Principal Scientist</td>
<td>VTT, BA2502</td>
<td>T4.1</td>
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</tr>
<tr>
<td>Nieminen Anna</td>
<td>Project manager, Research scientist</td>
<td>VTT, BA2502</td>
<td>T3.2, T3.3, T5.1</td>
<td>3.3</td>
</tr>
<tr>
<td>Rossi Jukka</td>
<td>Senior scientist</td>
<td>VTT, BA2502</td>
<td>T4.1</td>
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</tr>
<tr>
<td>Sevón Tuomo</td>
<td>Senior scientist</td>
<td>VTT, BA2502</td>
<td>T1.1, T1.2</td>
<td>2.6</td>
</tr>
<tr>
<td>Strandberg Magnus</td>
<td>Research trainee</td>
<td>VTT, BA2502</td>
<td>T3.1, T3.2</td>
<td>3.5</td>
</tr>
<tr>
<td>Taivassalo Veikko</td>
<td>Senior scientist</td>
<td>VTT, BA2502</td>
<td>T2.1, T3.3</td>
<td>2.5</td>
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<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>13.2</strong></td>
</tr>
</tbody>
</table>
5. Risk management

The project work essentially consists of numerical simulation work which means that generally the risks are relatively small compared to e.g. experimental activities. The greatest risks are associated to the availability of personnel and possible problems with the functionality and availability of the software used in the project. At the moment, there are only few people who master the area of severe accident as a whole. Special know-how on certain phenomena and specialized codes is focused to individual persons. Because of this, the individual contribution of key personnel cannot be easily replaced in case the key person for a specific task is not available. This highlights the importance of training and sharing experiences.

It also has to be taken into account that realisation of some international projects is still uncertain since decisions of financing are not ready when this application is submitted. In case of the realisation of such external risk, the possibility to modify the project plan accordingly will be discussed.

Because the project mostly consists of work packages and tasks which are not dependent on each other, the realisation of a risk in one task is not foreseen to have effect on the progress of the other tasks.
References


EURATOM/2013/59. Council directive laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation.


### Comprehensive Analysis of Severe Accidents

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
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<td>Volume</td>
<td>Personnel</td>
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<tr>
<td>T3.2 Pool scrubbing</td>
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<td>T3.3 OECD/NEA THAI-3</td>
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<tr>
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<tr>
<td>TOTAL</td>
<td>13.2</td>
<td>157</td>
</tr>
</tbody>
</table>

### Comments

**Travel:**

- T1.1 Participation to two OECD/NEA BSAF meetings (2 x 2 k€) and to ERMSAR2017 seminar (1.5 k€, abstract submitted)
- T1.2 Participation to CSARP/MCAP meeting
- T3.2 Participation to ERMSAR2017 seminar (1.5 k€, abstract submitted)
- T3.3 Participation to OECD/NEA THAI-3 PRG3 and PRG4 meetings (2 x 1.5 k€)

**Other:**

- Two participation fees applied to be funded 100 % by VYR:
  1) T1.2 CSARP (35 kUSD/a ~ 31 k€)
  2) T3.3 OECD/NEA THAI-3 participation fee is in total 47.5 k€ and the project lasts for 3.5 years
    - The contributions will be apportioned equally over the project duration
    - In 2017 2/7 of the participation fee will be paid similarly as in 2016
SAFIR Project plan

CATFIS
Chemistry and Transport of Fission Products

Teemu Kärkelä,
Project manager
VTT Technical Research Centre of Finland Ltd

Tommi Kekki,
Deputy project manager
VTT Technical Research Centre of Finland Ltd
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1. Research theme and motivation

In this CATFIS project both the previous TRAFI and FISKES projects of the SAFIR2014 program are combined. Both projects were focused on studies of fission product chemistry in a severe nuclear power plant accident. The duration of CATFIS project is four years (2015-2018). The goal of CATFIS project is to reduce the uncertainties associated with estimating the potential release of radiotoxic FPs. The formation of air radiolysis products and their impact on the fission product speciation will be investigated. The research efforts are concentrated especially on the chemistry and transport of iodine and ruthenium, which were defined as high priority issues in SARNET evaluation of Severe Accident Research Priorities. Another important topic is to find out the effect of pool scrubbing phenomenon in the containment pools or in the filtered containment venting systems (FCVS) on the retention of fission products.

The second goal of CATFIS project is to bring the experimental data to models. The work is focused on utilizing the gathered data in deriving mathematical models e.g. on phenomena which are not considered in the current models at all. And as an ultimate goal, the models will be implemented to severe accident analysis codes. Previously, ChemPool software has successfully been coupled with MELCOR input and result files to calculate the equilibrium composition and the pH of the water pools. The chemistry model of ChemPool needs to be kept updated.

The contribution of SAFIR2014 program and VTT on the European research has been significant. VTT has coordinated e.g. the source term work package in EU SARNET2 network. During 2013, the EU SARNET network was combined with the EU NUGENIA network. Currently, VTT is a member in the coordination team of the NUGENIA Technical Area 2.4, which is dedicated on source term studies. Furthermore, we have also presented the latest results on the chemistry of iodine and ruthenium in the 6th and 7th European Review Meetings on Severe Accident Research (ERMSAR 2013 and ERMSAR 2015). The main findings of VTT’s studies have also been presented in several scientific publications. VTT’s high level of expertise on iodine and ruthenium source term studies has also been recognized internationally and VTT was invited to participate in the scientific committee of the Iodine Workshop 2015 as well as to give two talks on the main outcomes of FP chemistry studies, which were performed under SAFIR programmes. VTT was also co-organizing the NUGENIA TA2.4 Source Term workshop 2015 and VTT was invited to give a presentation on FP transport phenomena. These workshops gathered the latest knowledge on the chemistry and transport of iodine and ruthenium during a severe accident. As an outcome of the workshops, the future research needs on this area were defined. The international collaboration will be further developed in CATFIS project, e.g. through EU NUGENIA, NKS and OECD/NEA networks as described below. The obtained results in CATFIS experiments will be shared with e.g. the members of EU NUGENIA network and at the same time the latest knowledge in source term area on European level will be provided to SAFIR2018 members. The main findings will be presented in international conferences on the field of severe accident research. As an outcome of the previous work performed in the CATFIS and TRAFI projects under the SAFIR2014/2018 programmes, VTT is starting a direct collaboration with JAEA (Japan) and GRS (Germany) on the high temperature chemistry of iodine and on the validation of FP transport models, respectively.

The results of the CATFIS project will also be summarized in scientific publications and in PhD theses. Additionally, the educational aspect will also be addressed by employing an MSc student to perform some of the experiments and to introduce the student to the European research community (e.g. NUGENIA network) on severe accidents.
1.1 Background and state-of-the-art

When fission products (FPs) are released from the nuclear fuel and they are transported into a primary circuit, in which part of the FPs are adsorbed/deposited on the surface of primary circuit. As an example, when the transport of ruthenium in air atmosphere was simulated in CHEMPC project (SAFIR2010 program), the deposition of RuO$_4$ on the surfaces was found to be diffusion limited reactive condensation, as the surface temperature decreased below 800 °C [Kärkelä et al., 2014]. A change in the RCS conditions may lead to a release of fission products from the surfaces of primary circuit even after a long time. In these tests most of the ruthenium transported through the model primary circuit as RuO$_2$ particles, whereas the gaseous RuO$_4$ fraction ranged up to 5 %. The gaseous fraction was increased by the revaporation of Ru from the deposit on the surfaces. In case of a severe accident, the main air radiolysis products are ozone (O$_3$) and nitrogen oxide species NO$_2$ and N$_2$O, which can be considered as oxidizing compounds [Mun et al., 2006]. As NO$_2$ reacts with water it forms nitric acid (HNO$_3$). In TRAFI and CATFIS projects during 2014-2015 (SAFIR2014 and SAFIR 2018 programmes) it was observed that the studied air radiolysis products (N$_2$O, NO$_2$, HNO$_3$) increased the transport of ruthenium through a model primary circuit [Kajan et al., 2016]. These experiments were and are the current state-of-the-art on the ruthenium transport studies internationally. The most alerting result was the effect of gaseous NO$_2$ to be able to oxidize the lower oxides of ruthenium to gaseous RuO$_4$. The increase in the gaseous Ru fraction was at least an order of magnitude when compared with the previous studies at pure air atmosphere. In addition, the gas flow through the primary circuit in an accident would also contain a mixture of FP aerosols. For example, a high fraction of iodine is expected to be released as CsI particles from the primary circuit to the containment atmosphere, especially at reducing conditions [Herranz et al., 2015]. In CATFIS (2015) it was observed that the transport of ruthenium in gaseous form increased significantly due to the airborne CsI in the model primary circuit [Kärkelä et al., 2016], the transport was higher than due to air radiolysis products. This observation has again a notable impact on the current state-of-the-art knowledge. At the moment the reason for this is not yet clear, the formation of a gaseous ruthenium-iodine compound is suggested. This work was conducted at VTT as a part of NKS collaboration with Chalmers University of Technology. Currently, the effect of fission product aerosols or air radiolysis products on the transport of Ru at air ingress conditions is poorly known. Most of the international experimental programmes on the Ru chemistry in the primary circuit, such as the OECD/NEA STEM/START programme [Clément and Simondi-Teisseire, 2010], have mainly been focused on studies at pure air or steam atmosphere. At the moment, these VTT-Chalmers separate effect experiments on the ruthenium transport are the only ones performed considering also the effect of air radiolysis products and fission products. The effect of the main air radiolysis products and FP aerosols on the transport of ruthenium should be investigated in detail. That would give more realistic information on the behaviour of Ru than what is currently available. These results gathered under SAFIR programs have been noticed also internationally and the main observations/experiments will repeated in the ongoing OECD/NEA STEM-2 program (years 2016-2019). Also the scope of STEM-2 has been changed and now the effect of air radiolysis products on the transport of Ru has been raised as one of the priority topics. Additional information on STEM-2 program is given below.

Concerning studies on iodine chemistry in the primary circuit, it is typically assumed that caesium iodide is the main iodine compound formed in the reactor coolant system. This assumption leads to a low release of gaseous iodine into the containment, because the current integral codes do not take into consideration the effect of chemical reactions on the primary circuit surfaces. Also, the previous studies have mainly focused on the reactions taking place in the gas phase [Gouëllo et al., 2013; Grégoire et al., 2012]. However, the importance of surface reactions as a source of volatile iodine is even increased at the late phase of accident when the thermal/hydraulic conditions of the circuit are changing. The effect of surface reactions on the release of iodine from CsI deposit at 650 °C was studied in CHEMPC project (SAFIR2010 program). It was found out that in certain conditions the fraction of gaseous iodine released was very high. Increasing hydrogen concentration decreased the transport of gaseous iodine. This was also verified by IRSN in their ongoing studies with CHIP facility on the gas phase reactions of iodine in primary circuit conditions [Herranz et al., 2015]. As boron is used as a neutron absorber in the control rods or in the solvent of coolant water to control the fission reaction, its effect on the CsI chemistry was studied with experiments in CHEMPC project. Generally, the effect of surface reactions of boron on the release of volatile iodine is poorly known. In the experiments it was noticed that CsI and B$_2$O$_3$ reacted forming a solid, glassy compound with characteristic bands of CsB$_2$O$_3$-v [Sinclair et al., 2006] in the Raman spectrum. During 2011-2013 in TRAFI project (SAFIR2014 program) it was found out that iodine was mainly released in gaseous form from solid CsI precursor even at a relatively low temperature of 400 °C. When molybdenum or boron was mixed with CsI, they reacted with caesium and thus the release of gaseous iodine was enhanced. The concentration of gaseous iodine transported through the primary circuit into the containment atmosphere during the degradation phase was also the highest in the Phébus FPT3 test [Girault et al., 2013] when compared to other tests of Phébus
program. FPT3 test was the only test performed with a boron carbide (B$_2$C) control rod. The high gaseous iodine fraction likely originated from the reactions of the degrading control rod with fission products. Considering the use of boron as a neutron absorber, which can be e.g. dissolved in the coolant water of PWR primary circuit, it is important to understand the chemistry of CsI-B system in the circuit. The importance of the boron effect is also high internationally; Japanese Atomic Energy Agency (JAEA) has contacted VTT on this matter and proposed collaboration to investigate the impact of boron on the possible source term. The collaboration will be started in October 2016. From previous TGA/DTA analyses, the temperature at which the reaction of caesium and boron is initiated has been defined for a specific B/Cs ratio in CATFIS 2015. Relation between the formation of Cs-B glassy compound and the release of gaseous iodine has been suggested. Then, decreasing the temperature below the formation temperature of the glass would confirm this link, if, as expected, the amount of gaseous iodine is decreased for the same B/Cs ratio. Finnish utilities have required data on air ingress, since such accident may take place during refuelling. Concerning the air ingress topic, most of the FP release data come from the experiments performed by AECL on CANDU fuels, which are not representative of LWR fuels. These experiments do not cover the FP transport issue, thus the reaction under air as well as in oxidizing and reducing conditions in general should be investigated. VTT’s previous experiments showed that the amount of gaseous iodine in Ar/Air is lower than in Ar/H$_2$O, this would mean that the formation of gaseous iodine, i.e. formation of caesium borate, is affected by the presence of water, more than a high oxygen potential. The possible influence of Cs to I molar ratio higher than one on the release of gaseous iodine was investigated with few tests in 2014-2016. The higher fraction of Cs in the precursor seemed to decrease the transport of gaseous iodine by forming CsI particles, even at 400 °C reaction temperature. These findings are the state-of-the-art of iodine chemistry taking place on the surface of primary circuit. In future, more information on the formation of the caesium molybdates and on the difference of the fraction of gaseous iodine released according to the initial Mo/Cs ratio is needed. It is expected that the presence of air would likely alter the reaction between molybdenum and caesium iodide in the primary circuit.

Besides the reactions of iodine inside the primary circuit, it is important to find out how iodine reacts with various materials in containment. Iodine may exist in the atmosphere of containment in a gaseous form and as fine particles. Gaseous iodine will react with air radiolysis products and form fine aerosol particles. Previously, the reaction kinetics of inorganic and organic iodine species with air radiolysis products was studied with EXSI-CONT facility during CHEMPC project. During 2012 a model for reaction between elemental iodine and ozone was derived at 120 °C. The model covers the concentrations of elemental iodine (I$_2$) and ozone (O$_3$) ranging from 0 to 10 ppm and 0 to 4000 ppm respectively. During 2010 VTT participated in study at Chalmers University of technology on the oxidation of organic iodine by gamma radiation. On the basis of the experiments it can be said that the oxidation of iodine depends on the type of radiation used. In most international experimental programmes on the radiolytic oxidation of iodine the source of radiation has been $^{60}$Co (gamma rays). However, during a severe accident the most important dose in the containment atmosphere is coming from beta radiation [Bosland et al., 2010; Penttilä et al., 2013]. At the moment it is known that beta radiation is much more effective in creating iodine oxide particles. During 2012 a new BESSEL facility, which can be used to study the effect of beta and alpha radiation, has been built in TRAFI project. As a part of first experiments in TRAFI project, the formation of particles by beta radiation was observed [Kärkelä and Auvinen, 2013]. Further data analysis suggested that an equilibrium was reached between gas phase iodine compounds and iodine species deposited on wall surfaces of the facility [Kärkelä et al., 2015]. Similar observation on the constant concentration of iodine in the gas phase was also reported in the large-scale Phébus FP tests, however the reason for that remained unknown. In the experiments of TRAFI project, the rate of new particle formation at that equilibrium was low. When the facility was purged with oxygen, a new formation of particles was observed in every experiment. It suggested that the radiolysis reaction products were limiting the particle formation. In order to develop models for the radiolytical oxidation of iodine by beta radiation, the quantification of the reaction rates needs to be done.

In a severe accident, nitric acid (HNO$_3$) is formed in the steam-rich air or N$_2$ containment atmosphere as a result of irradiation by the airborne radioactive fission products. Nitric acid can also be formed in the containment water pools, in which N$_2$ has dissolved into sump water and is exposed to radiation originating from the fission products trapped in the sump. The pH value of the containment pool decreases with the increasing acid formation and thus leads to a possible release of iodine from the pool to the containment atmosphere. The formation of nitric acid by irradiation has previously been studied at Oak Ridge National Laboratory (ORNL) [Beahm et al., 1992]. Those studies were conducted with gamma radiation (Co). However, only a few further studies have been performed although the formulas presented in NUREG/CR-5950 report [Beahm et al., 1992] are applied in the plant evaluation and also in the estimation of pool pH value. The previous study of ORNL does not address the production of HNO$_3$ in all severe accident conditions, for example the effect of beta radiation is unknown. MELCOR/ORIGEN calculations estimate that the dose rate in the containment atmosphere is mainly originated
from beta radiation. The contribution of gamma radiation on the dose rate is roughly an order of magnitude lower. Thus the effect of beta decay on nitric acid formation needs to be studied. The additional HNO$_3$ impacts on the pH of containment water pools and thus it should also be taken into account in iodine source term studies. VTT performed experiments in CATFIS 2015 project on the HNO$_3$ formation by beta radiation in humid air. The results indicated that a radiation dose of four orders of magnitude lower by beta than gamma radiation generates a similar amount of HNO$_3$ molecules. This is a significant difference in the HNO$_3$ yield and it verifies the importance of this study. The dependence of HNO$_3$ formation on the beta radiation dose rate was also observed. On the basis of these observations, the accumulation and formation of HNO$_3$ in the containment pool due to beta radiation in the atmosphere, or directly in the pool, should be experimentally investigated in order to be able to give more reliable estimations e.g. on the iodine source term.

When fission products are transported into the containment atmosphere it is of the utmost importance to trap the airborne fission products before they are released into the environment. One method is to utilize the water pools in the containment building or in the filtered containment venting systems (FCVS) and direct the flow of fission product compounds through the pools. The pools are rather efficient in trapping particles and some gaseous compounds, although the gas bubbles can transport a notable fraction of the fission products through the pools back into the containment atmosphere, or in a worst case into the environment. There are some unknown features related to the pool scrubbing phenomenon [Jacquemain et al., 2016]. One of the internationally recognized open questions is the long-term behaviour of the pool. In the course of the accident evolution, the pool will get loaded with fission products and other materials, such as structural materials. The question is how these materials will effect on the trapping efficiency of the pool, is it possible that the pool will act more like a source of fission products. Another open question is the behaviour of the constantly problematic organic iodine. Its behaviour in the pool should be investigated in a long-term. Additionally, the existing experiments have usually been performed at low temperatures, such as at 20 °C water temperature. The research should be extended to cover also realistic pool chemistry and pool temperatures at boiling point or close to it. Also the effect of heat and radiation in oxygen rich or low pool conditions need to be examined to verify the possible decrease in the pool trapping efficiency for fission products. Radiation and heat together with oxygen may change the alkaline chemistry of the pool.

The previous CHEMPC (SAFIR2010 program) and TRAFI (SAFIR2014 program) projects have produced a wide database on iodine chemistry in a severe accident. The studied phenomena have focused on topics which have been poorly known. For example, the interest has been on the release of iodine from deposits on the primary circuit surfaces. The FP deposits on the circuit surfaces are an important source of volatile iodine, especially in the late phase of accident. Another important topic has been the oxidation of gaseous iodine at containment conditions. It has been studied by exposing gaseous inorganic iodine and organic iodine to ozone and to UV, beta and gamma irradiation. Oxidation of iodine can change the chemical form of iodine from gaseous compound to solid particle, thus it also has a strong impact on the possible iodine source term. The behaviour of iodine containing particles has been studied by depositing radiolabelled iodine oxide and caesium iodide particles on containment surface materials, such as aged paint and metals. The speciation of deposits was analysed and in the next phase the particle samples were exposed to heat, humidity and gamma irradiation. The experiments indicated that iodine was separated from the deposits and then iodine either reacted with the substrate or it was directly released to the gas phase. As these phenomena are weakly understood internationally, their current mathematical models in the severe accident analysis codes are as good as the phenomena are known or the phenomena have not been considered at all. Thus, the gathered data needs to be utilized in deriving new mathematical models. And as an ultimate goal, the models should be implemented to severe accident analysis codes.

MELCOR ("melting core") is a fully integrated, engineering-level computer code by Sandia National Laboratories whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants. But MELCOR code is not able to consider aqueous species and how they affect the pH in the containment’s pools. MELCOR also contains an iodine transport model, but when tested by VTT it was found to be unusable. ChemPool software is able to interpret the MELCOR input file to automatically setup the network of control volumes inside the containment. After MELCOR simulation the results from MELCOR are exported to text files containing temperatures, pressures and compositions of the pools at each time steps. The estimated formation rates of acids and dissolved salts from fission products are then coupled with these and equilibrium compositions and pH values of the pools can be calculated with ChemPool at each time steps. Previously, ChemPool has successfully been coupled with MELCOR input and result files to calculate the equilibrium composition and the pH of the water pools. Reactor safety studies have shown that of all quantities affecting iodine volatilization, the pH of principal water pools is the most important. For values of pH > 7, the non-volatile forms I(-a) and IO$_2$(-a) are found...
1.2 Objectives and expected results

The main objective in CATFIS project (2015-2018) is to find out the mechanisms how gaseous fission products, especially iodine and ruthenium, are released from primary circuit into containment. In addition, the behaviour of iodine in the containment gas phase will be studied. Another objective is to find out the effect of radiation products, especially iodine and ruthenium, are released from primary circuit into containment. In addition, the behaviour of iodine in the containment gas phase will be studied.

Almost exclusively. However, as pH drops the creation of molecular iodine (I₂) begins. For pH < 3, I₂ is the dominant form. This phenomenon varies somewhat with temperature, concentration, and radiation dose, although these are secondary effects [Weber and Beahm, 1999]. Thus controlling the pH in the containment is important as in acidic conditions radioactive iodine could be formed and released into atmosphere. Currently the nitric acid (HNO₃) formation rate in ChemPool is based on NUREG report [Beahm et al., 1992] that uses gamma radiation only. However, MELCOR/ORIGEN calculations estimate that the dose rate in the containment atmosphere is produced mostly from beta radiation. Gamma dose rate is lower roughly with an order of magnitude. This could mean that the total formation rate of HNO₃ is higher than previously anticipated. HNO₃ formation rate in ChemPool needs to be updated for beta radiation and its effect on the pH of the water pools needs to be estimated by recalculations on known severe accident scenarios. As part of CATFIS 2015 project, the first calculations using the data of HNO₃ formation in humid air by beta radiation (experiments performed in CATFIS project) were done. The results indicated that the amount of NaOH needed for the pH control of water pools can be higher than previously expected. These observations will be verified both experimentally and by ChemPool calculations in 2016. On the basis of ChemPool development a more realistic estimation on the sump acidification of a Finnish NPP in a hypothetical SA can be given. The HNO₃ formation model needs to be updated with the data covering also the formation by beta radiation in pool. Key finding of the Phébus FP program [Raimond et al., 2013] was that the evidence from the Phébus experiments indicated that caesium molybdate (Cs₂MoO₄) was the dominant chemical form of released Cs (instead of CsOH). Caesium molybdate is highly soluble to water just like caesium hydroxide, but unlike caesium hydroxide which is a strong base, it does not affect the pH. To estimate the iodine chemistry in pool more accurately, the chemistry libraries of ChemPool need to be continuously updated as new data become available.

The international co-operation was very active during CHEMPC project and that was continued in TRAFI project. In the framework of International Source Term Program (ISTP) a sampling system for the CHIP facility at IRSN Cadarache was constructed at VTT in 2007. VTT participated in the testing of the facility in 2008 at the Cadarache research Centre. VTT has followed up the results of Phébus FP experimental program and has participated in the review of e.g. the final reports for FPT-2 and FPT-3 experiments. In addition, VTT has also participated in the aerosol measurements and the interpretation of the results of the international ARTIST and ARTIST2 programs on aerosol and droplet transport in the steam generator of a pressurized water reactor during postulated steam generator tube rupture (SGTR) conditions. The resuspension of fine particles in turbulent tube flow has been studied with experiments at VTT as a part of ARTIST program. As a result, a master’s thesis was completed in the beginning of 2008. As a part of ARTIST2 program the deposition of particles on heat exchanger surfaces was measured in condensing conditions at the end of 2009. During TRAFI project VTT has also collaborated with PSI to investigate the deposition of aerosol particles in turbulent natural convection flow, which is generated to the containment building due to the temperature difference between various surfaces. The results of this extensive study were summarized in a form of PhD thesis during summer 2015. In addition, VTT participated in the work of evaluation group of ISTC EVAN program. The results from the various studies have been distributed, for example through SARNET2 network, to be utilized in the development of models estimating the transport of fission products. During 2011-2012 the behaviour of iodine oxide and caesium iodide particles on containment surfaces (e.g. paint and metals) and the desorption of iodine back to the gas phase has been studied in collaboration with Chalmers University of Technology as a part of NKS-R program. The collaboration has been continued during 2014-2015 with a focus on the high temperature chemistry of ruthenium. Several publications and conference presentations e.g. on the iodine research, on the behaviour of ruthenium and on aerosol resuspension were published as an outcome of the international co-operation. In addition, VTT also participated in writing of the OECD State-of-the-Art report on aerosols in nuclear safety published in 2009. Since 2011 TRAFI project has participated in the follow-up of OECD/NEA STEM program and the results of VTT’s ruthenium and iodine experiments have been presented to the partners. The main focus in the program has been on the behaviour of iodine containing aerosol in the containment building and on the transport of Ru through the RCS into the containment building. The STEM program ended in June 2015. The follow-up is continued in a new program STEM-2 (2016-2019), see below. In addition, the follow-up of OECD/NEA BIP-3 program was initiated in 2016 as described below. VTT is currently strongly involved into the work of NUGENIA TA2.4 and has a membership in the coordination team of it.

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The main objective in CATFIS project (2015-2018) is to find out the mechanisms how gaseous fission products, especially iodine and ruthenium, are released from primary circuit into containment. In addition, the behaviour of iodine in the containment gas phase will be studied. Another objective is to find out the effect of radiation...
on the formation of air radiolysis products and their impact on the fission product speciation. Also the retention of fission products in the containment pool or FCVS scrubber due to the pool scrubbing phenomenon will be investigated. The data gathered will be utilized to develop mathematical models on fission product transport. The international collaboration will be further strengthened through the NKS-R collaboration with Chalmers University of Technology and through a direct collaboration with JAEN and IRSN. The international collaboration will be carried out by participating in OECD/NEA STEM-2 and BIP-3 programs and also in NUGENIA TA2.4 area international projects.

The project is divided into five workpackages in 2017. The first workpackage is focused on the primary circuit chemistry of iodine. General objective of this work is to identify if chemical reactions and revaporisation process at the primary circuit surfaces may have a significant effect on the physical and chemical form and on the transport of fission products. In the proposed work, the impact of initial Mo/Cs molar ratios on the formation of gaseous iodine in the circuit will be examined. In PWR fuel, the molar inventory of molybdenum is higher than the one of caesium by a factor of 1.5 to 2. Nevertheless, the Mo/Cs molar ratio in the primary circuit is highly dependent on the accident sequence and can vary over a larger spectrum during the transient. The precursor samples will be exposed to Ar/Air and Ar/H_2O atmospheres at 650°C and the reaction products will be analysed in detail. The studies will give more information on the formation of the caesium molybdates and on the difference of the fraction of gaseous iodine released according to the initial Mo/Cs ratio. The chemical reactions of fission product deposits taking place on the primary circuit surfaces are not considered in the current SA analysis codes, although the deposits can act as a significant source of volatile iodine especially at the late phase of accident when e.g. thermal-hydraulic conditions are changing. By the end of 2018, the objective is to utilize the previous and new experimental data in developing models for the release of gaseous iodine from the FP deposits on RCS surfaces and for the subsequent iodine transport, as the phenomena has been studied in TRAFI and CATFIS projects for several years. It will take place as a visit at IRSN and as collaboration with their personnel to implement the models into the ASTEC/SOPHAEROS module. The collaboration with IRSN will be more active than before and several publications are expected to be finalized from this task. These new models and code developments will enable the consideration of phenomena, which were previously poorly-known, in the analysis of severe accident.

The formation of nitric acid by beta radiation in the containment gas phase and sump will be studied in the second workpackage. Previous studies on nitric acid formation have been conducted with gamma radiation. As mentioned above, in a severe accident the most important source of ionizing radiation is beta radiation. However, the rate of nitric acid formation due to beta radiation may be a lot higher than expected. The objective is to find out the formation rate of HNO_3 by beta radiation (e.g. P-32) in humid air or nitrogen atmosphere. The concentration of water, oxygen/nitrogen ratio, dose rate and radiation dose will be varied. The results will be compared with the previous tests conducted with gamma radiation. The understanding on these objectives has increased due to experiments in 2015-2016. By the end of 2018, the formation rate will be defined in a temperature range from 20 to 120 °C. Another objective is to extend the experimental work to cover mixtures of water pool and air or N_2 atmosphere as well, in order to gather information on the resulting HNO_3 content in the water pool. The radiation source can be located in the gas phase or in the liquid phase. These studies will also give information on the formation of air radiolysis products in general and on their thermodynamic equilibrium with other species. The effect of the formed HNO_3 on the sump acidification and on the possible release of iodine from the sump will also be considered, especially in the third workpackage (see below). This kind of information is needed to enhance the nuclear safety. The results will be summarized in a scientific publication, which will be part of the PhD thesis of Teemu Kärkelä (VTT).

The third workpackage is focused on the development of ChemPool software. The objective is to improve the chemistry model. That will include the updating of nitric acid formation rate to notice the impact of beta radiation as well. The data for it will be received from the experiments of WP2. Secondly, the chemistry model of Chempool will be updated continuously during the project in 2015-2018. The third objective is to recalculate Olkiluoto and Lovisa severe accident scenarios with the updated chemistry model and HNO_3 formation rate. This task is important in the course of project, since it will give an overall view on the effect of various phenomena to pool pH and to speciation of iodine concerning especially Finnish NPPs and thus this knowledge will increase the nuclear safety. The results will be summarized in a scientific publication.

The objective in the fourth workpackage is to investigate the retention of fission products due to the pool scrubbing phenomenon. It can take place e.g. in the containment pool or FCVS scrubber. The objective is to focus on the long-term behaviour of the pool as a trap for the fission products. As a result, the trapping efficiency of the pool when it has filled-up with fission products and other structural materials will be found out. Another objec-
tive is to investigate the behaviour of organic iodine in the pool. As organic iodine is known to be difficult to trap, the effect of its volatile nature on the release from the pool will be verified. As a third objective, also the retention of aerosol particles will be studied. The main emphasis in this workpackage is to understand how the pool behaves in a long-term. And as another main factor of this WP, the pool temperature will be varied from 20 to 100 °C in the experiments. Also the pool chemistry will be as realistic as possible in the experiments corresponding to Finnish NPPs. It is possible that when the pool with some oxygen content is exposed to heat and radiation it results in a change in the alkaline chemistry of the pool, which may decrease the retention of fission products in the pool significantly. This needs to be studied with experiments. Additionally, several other pool parameters will be varied in the experiments, see Chapter 2. The work will be initiated in 2017 with a focus on the needs of Finnish nuclear power plants. VTT will get technical advice from PSI (Switzerland), which has performed pool scrubbing experiments previously and thus PSI has also knowledge on the hydrodynamics of the pool. Additionally, VTT participates in the NUGENIA TA2.4 area project IPRESCA, which begins in 2017. The IPRESCA project summarizes the current knowledge on the pool scrubbing phenomenon internationally. The experimental data of WP4 will be also utilized in the SAFIR RG2 CASA project, in which the experiments will be simulated with the MELCOR and ASTEC codes and the calculation results of the codes will be compared. The gathered knowledge in the VTT’s pool scrubbing experiments and modelling will also be shared through OECD/NEA THAI-3 program. The follow-up of THAI-3 is being performed in the SAFIR2018 RG2 CASA project. The publications of the results from this task are part of the PhD thesis of Teemu Kärkelä (VTT).

The follow-up of OECD/NEA STEM-2 (Source Term Evaluation and Mitigation Issues) and OECD/NEA BIP-3 (Behaviour of Iodine) programmes, which both began in January 2016, will be continued in the fifth workpackage. The four-year STEM-2 programme is focused on the transport of ruthenium in the primary circuit conditions and on the reactions of particulate iodine on the painted and metal containment surfaces. The three-year BIP-3 programme is focused on the behaviour of gaseous inorganic and organic iodine on the painted containment surfaces, especially the adsorption and desorption phenomena. The specific project plans of both programmes were distributed to SAFIR2018 RG2 members in summer 2015. Due to the VTT’s unique experiments on the Ru chemistry in RCS (SAFIR2014 and SAFIR2018 programs), VTT’s technical advice and expertise has been asked in the planning of experiments in OECD/NEA STEM-2 program. Another area of VTT’s expertise will be needed on tests related to the release of gaseous iodine from iodine containing aerosol deposits on containment surfaces. As described above, VTT has previously successfully performed similar experiments, under SAFIR2014 program as well, and the results have been taken into consideration in the planning of STEM-2. Both OECD programmes produce valuable data on the behaviour of fission products, especially iodine and ruthenium, which is needed in the estimation of possible source term.

1.3 Exploitation of the results

The experimental results produced in CATFIS project (2015-2018) will increase the knowledge on the behaviour of fission products during a severe accident. CATFIS is also currently the only project, in which the formation of air radiolysis products and their effect on the transport of FPs has been investigated. As it has already been shown in TRAFI and CATFIS projects, the oxidative radiolysis products can enhance the transport of FPs. The experimental data can be used e.g. by the nuclear power companies and the radiation and nuclear safety authority for PSA level 2 analysis of the existing nuclear power plants. A complete experimental database will be finalized by the end of 2018.

Revaporation of volatile fission products has been evidenced in every recent experiment (Phebus FPT2, Phebus FPT3 and VERDON 2), in which the transport of fission products within primary circuit has been studied using irradiated fuel samples. Deposition and revaporation of fission products takes place, whenever the temperature of the circuit, flow rate, flow composition or release rate from the core changes. The influence of these processes on the FP transport is very significant. Based on VTT’s analysis of Phebus FPT1 test [Auvinen and Jokiniemi, 2003], the deposition at inlet of the steam generator was quantified in Phebus FPT3 experiment. It was found out that approximately one third of caesium core inventory was located in that section at the end of the test. Similar quantification can’t be done for iodine, because the radioactive iodine has decayed long before samples can be retrieved from the experimental facilities. We believe though that the timing of iodine release, its release fraction as well as speciation depend very significantly on the chemical reactions taking place on the primary circuit surfaces. The effect of these reactions has only been studied in SAFIR program, which can provide now essential experimental data for the nuclear safety community.
The ASTEC (Accident Source Term Evaluation Code) was developed in 1995 jointly by the IRSN and its German counterpart, GRS, for severe accident analysis and the code simulates the overall nuclear power plant response. The development of the code played a main role in the work of the previous EU SARNET network and the work is continued in EU NUGENIA network by integrating models based on the knowledge of the network. SOPHAEROS, module of the ASTEC, computes chemistry and transport of FPs in the primary circuit during a severe accident [Cousin et al., 2013]. Until now, reactions involving condensed species were considered as minor and weren’t taken into account in the models. However, it has recently been shown by VTT that the surface reactions can have a notable influence on the source term (e.g. iodine) when the circuit conditions are changing [Kalilainen et al., 2014]. The exploitation of the experimental results is directly related to the validation of the ASTEC code and to the implementation of observed phenomena. This part of the work will be carried out in collaboration with IRSN in 2018. It will improve the accuracy of source term evaluations and enables a better understanding on FP speciation. Thus it will also improve the source term mitigation measures. Although ASTEC has become the European reference software, end users are international.

![Simplified Reaction Scheme of iodine behaviour within the containment](image)

Recent experimental findings have changed the view how iodine behaves in the containment during a severe accident. VTT co-ordinated Source Term work package in the previous EU SARNET2 project, in which the following scheme for iodine reactions was derived (Figure 1). In addition to TRAFI and CHEMPC projects (SAFIR2010 and SAFIR2014 programs) data from Phebus FP, ISTP EPICUR, OECD BIP and BIP2, OECD STEM, OECD THAI, PSI Radiolysis tests and NKS NROI and AIAS projects are applied to quantify various reactions. However, TRAFI is the only project, in which the transformation of gaseous iodine to iodine oxides has been studied using representative beta radiation. Estimation of the iodine concentration in the containment atmosphere is not even possible, if any of the main processes (see Figure 1) remains unknown.

The behaviour of ruthenium in severe accident conditions has been estimated with calculations. However, the previous experiments conducted at VTT (Finland) and MTA EK (Hungary) have verified that the transport of radionuclides, gaseous Ru to the containment atmosphere can be higher than expected [Kärkelä et al., 2014]. The results of these studies have partially led to a launching of the ongoing OECD/NEA STEM/START program by IRSN [Clément and Simondi-Teisseire, 2010] in order to update the ruthenium transport models of ASTEC. VTT’s new experiments in TRAFI 2014 and CATFIS 2015 revealed the effect of FP aerosols and air radiolysis products on the transport of Ru. This information can’t be gained without experimental studies. These studies on more realistic conditions prevailing in the primary circuit benefit nuclear safety community as a whole by pointing out the gaps in the current knowledge on the behaviour of Ru. According to the results of the experiments in 2014 and 2015, the observed phenomena need to be considered in the severe accident analyses as well. These main observations will be repeated in the OECD NEA/STEM-2 program.

Even though most of the radiation dose in the containment building is originated from beta decay, the formation rate of nitric acid used in the severe accident analyses is based on gamma radiation. The proposed experiments
will result in more realistic formation rate of nitric acid using beta radiation as a source of ionizing radiation. The formation rate can be used in future accident analyses, thus also enabling the analysis of containment pH and iodine source term more exactly. The first ChemPool calculations based on the CATFIS experimental data have shown that the amount of NaOH needed to control pH can be higher than expected. This information benefits directly the power companies who are running the nuclear power plants. The development of ChemPool software with the latests experimental data will give a possibility to assess phenomena which are important considering a possible source term. The related severe accident scenario calculations are focused on Finnish NPPs.

The experimental evaluation of the retention of fission products in the containment pool or FCVS scrubber is of importance to ensure the mitigation of possible releases in a severe accident. The data on the pool scrubbing at more realistic conditions (high temperature, realistic chemistry) than the available data nowadays will be highly valuable for the power companies and authority in order to understand the long-term behaviour of the pool/scrubber and its effect on the trapping efficiency of FPs, such as organic iodine. The results have a direct impact on source term analysis. The first set of pool scrubbing data accommodating the features of Finnish NPPs will be available in 2017.

The development of mathematical models based on the experimental iodine data will lead to more accurate source term estimations. The phenomena, which previously have not been considered in the analyses at all, will be implemented as new models to severe accident analysis codes. This will promote a better understanding on the importance of various accident sequences and also on their impact on the possible source term. That will serve the nuclear safety community as a whole.

Results from OECD/NEA STEM-2 and BIP-3 programs increase the knowledge on the behaviour of both gaseous and particulate iodine on the painted surfaces in containment and on the primary circuit chemistry of ruthenium. These experiments complement the data from VTT’s tests carried out in SAFIR program. Previous studies at VTT on ruthenium chemistry increased the knowledge on the behaviour of ruthenium in air ingress accidents. Such accident could take place during the plant shutdown, when the reactor pressure vessel is open to the containment atmosphere. Experiments at VTT provide data on ruthenium transport and speciation, whereas experiments performed at CEA, EDF and MTA provide data on the release of ruthenium and other FPs. Experiments at IRSN as well as at Chalmers provide information on ruthenium chemistry in the containment building. The data is needed for PSA level 2 analysis of the existing nuclear power plants. The results from the STEM-2 and BIP-3 programs will also be valuable in the interpretation work on Phebus FP experimental results.

The results of the work in CATFIS project will be summarized in scientific publications as presented above. The main findings will also be presented in international conferences on the field of severe accident research. Since the international collaboration has been active during SAFIR2010 and SAFIR2014, it will be further enhanced in CATFIS project. As a first step, the results gained will be shared with EU NUGENIA network and also the knowledge on European level in the source term area will be provided to SAFIR2018 members. Further collaboration with other international organizations will also be pursued during the four-year period.

1.4 Appropriateness of the project to SAFIR2018 programme

In CATFIS project are combined the previous TRAFI and FISKES projects (SAFIR2014). The merits of previous work have been summarized in Chapter 1. Both previous projects were dedicated on the chemistry of fission products in a severe accident. TRAFI and FISKES projects were already in collaboration and as a result state-of-the-art experimental facilities have been built, e.g. BESSEL facility (Beta irradiation vESSEL), and several challenging experiments e.g. with radiotracers have previously been conducted together. The collaboration will be even closer as the experts of fission product chemistry are brought together in CATFIS project. A clear added value is that now the examination of more difficult FP chemistry phenomena (including e.g. radiation induced reactions of liquids, gases and aerosols at primary circuit and containment conditions) will be possible to conduct in CATFIS project. That leads to experiments and studies, which can also be considered as state-of-the-art internationally. As a special detail, the experiments in CATFIS project can be focused on specific needs of Finnish NPPs. This already verifies that CATFIS project is appropriate to SAFIR2018 program.

The understanding of fission product chemistry in a severe accident is also an important part of the scope of SAFIR2018 program. In order to be able to estimate the possible source term accurately, the chemistry and phenomena taking place in the accident situation need to be known in detail. For example, iodine speciation depends
on several factors when iodine is released from the fuel and transported through the circuit into the containment atmosphere. Therefore iodine chemistry has especially been taken into account in the framework plan of SAFIR2018 program (Annex 1), as well in CATFIS project. To investigate the fission product chemistry, it is important to consider the effect of radiation on the gaseous medium and also on the reactions of fission products. This point of view is considered in CATFIS project and the formation of radiolysis products and their reactions with FPs will be studied in 2015-2018. The radiolysis reactions have been noticed in the framework plan (Annex 1) as well. The retention of fission products in a water pool inside the containment building or in the FCVS scrubber need to be understood better, especially at the water boiling temperature or close to it. This so called “Pool Scrubbing” phenomenon is a research topic in CATFIS from the beginning of 2017 and it was also considered important in the “update 2017” of the framework plan of SAFIR2018 program (Annex 1). In case the water pool would not trap fission products efficiently and the pool would be e.g. in the FCVS scrubber, a potential risk of source term would be high.

The proposed experiments of CATFIS project will produce experimental data which will be implemented as models to severe accident analysis codes. That will enable to perform more accurate source term analyses by the end users. Also severe accident analysis work will be conducted with ChemPool software in the project. It will give plantspecific information (Finnish NPPs) on the pH in the containment pool and on iodine speciation. The data of CATFIS project will also be used as input for the ChemPool model updates and calculations. Thus, in practice all workpackages are connected to each other providing a compact project with state-of-the-art research topics.

CATFIS project educates young researchers to perform measurements and analyses on severe accident phenomena. As a result, at least two PhD theses are expected to be finalized by the end of SAFIR2018. The possibility to employ young research trainees to perform measurements as a special work and then to complete a MSc degree will be investigated along the four-year period. This serves also the purpose of SAFIR2018 program to educate new experts to the area of severe accident research.

International collaboration is very active in CATFIS project. The project has a membership in the coordination team of EU NUGENIA network Technical Area 2.4 (Source Term area). In addition, the project participated in the Scientific Committee of the Iodine Workshop (2015) considering the latest knowledge e.g. on iodine and ruthenium chemistry. The outcomes of this workshop have a strong impact on the future research lines on European level at least. The international collaboration includes also NKS-R collaboration with Chalmers University of Technology (Sweden) and the planned ASTEC development work will be performed in collaboration with IRSN (France). The interpretation of experimental results on the high temperature chemistry of iodine will be performed in collaboration with IRSN and especially with JAEA (Japan), which collaborates with VTT also in a separate project on the boron effect on iodine chemistry (starting in October 2016). CATFIS project has also a strong connection with several other organisations, like JRC-ITU and PSI (Switzerland). PSI will give technical advice to VTT in WP4 on the pool scrubbing related issues. WP4 is also connected to the NUGENIA TA2.4 ISPRESCA project, in which the knowledge of participants will be shared to summarize the current state-of-the-art in the pool scrubbing studies internationally. In addition, CATFIS project is also in collaboration with CASA project sharing the data of WP4 to be used in MELCOR and ASTEC simulations. And the experimental results of WP4 will be shared through the OECD/NEA THAI-3 project as well (SAFIR2018 RG2 CASA is performing the THAI-3 follow-up).

Further collaboration is being built continuously; VTT will be coordinating a task in the EU USTA project (starting in 2017 if accepted) on the uncertainties in source term analyses. The project has participants from 30 organisations all around the world, thus all the main contributors in the area of nuclear safety can be contacted easily. This promotes also the aim of SAFIR2018 program to increase the level of international collaboration and visibility.

To summarize, CATFIS project is suitable for SAFIR2018 programme. The scientific level of the project is high as it can be noticed from the amount of scientific publications published e.g. at least five scientific publications in 2015-2016 so far. The publications are both own and joint publications. The experimental results gathered in CATFIS project have also aroused attention internationally and on the basis of these merits VTT has been asked to participate in international projects and/or to coordinate some parts of the projects. CATFIS project has also an impact on nuclear safety. The above discussed experimental results and the experiments to be conducted give new information about the behaviour of fission products in a severe accident. The experiments have revealed phenomena which are not considered in the current severe accident simulation codes, e.g. the effect of reactions taking place on the primary circuit surfaces on iodine chemistry, nitric acid formation by beta radiation and the subsequent effect on the pH of containment pool. New experiments will produce information e.g. on the pool scrubbing phenomenon, which is an important method to trap fission products. The related model development on these phenomena will improve the nuclear safety when the models are implemented into severe accident analysis.
codes, such as MELCOR and ASTEC. The value for end users will come through the improved nuclear safety, e.g. when the observed results are introduced at the nuclear power plant in practice or when the gathered information are taken into consideration when the decisions on nuclear safety are done in future. In general, CATFIS project takes care of the maintenance and development of know-how on the experimental analysis of severe accident phenomena and on the development of models to describe the observed phenomena in collaboration with international organisations.

1.5 Education of experts

Young researcher MSc (Tech) Teemu Kärkelä participates in the project. He will be educated to conduct advanced aerosol measurements in the field of nuclear safety. The publications for example from the iodine experiments will be part of his PhD thesis. The dissertation is expected to take place by the end of SAFIR2018 program.

Nordic co-operation has been promoted by collaboration with Chalmers University of Technology and taking part in the NKS R-program. Previous experiments (TRAFI 2011-2012) on the reactions of iodine on different containment surfaces and iodine circulation in the containment atmosphere were part of PhD study of young researcher MSc Sabrina Tietze (Chalmers). Sabrina Tietze has participated in the experiments at VTT.

NKS-R collaboration with Chalmers University of Technology was continued in 2014-2015. Young researcher MSc Ivan Kajan (Chalmers) participated in the Ru experiments conducted at VTT in TRAFI and CATFIS projects. The focus in these studies was on the chemistry and transport of Ru in the primary circuit of nuclear power plant. The results of Ru experiments will be part of PhD theses of Teemu Kärkelä (VTT) and Ivan Kajan (Chalmers). The dissertation of Ivan Kajan will take place in October 2016. His PhD thesis includes two scientific joint-publications written on the basis of NKS collaboration in SAFIR programmes.

As a result of the previous TRAFI project (SAFIR2014), young researcher MSC (Tech) Jarmo Kalilainen finalized a PhD thesis on aerosol particles deposition in turbulent natural convection in June 2015. Jarmo Kalilainen visited also at PSI several times as a part of the PhD work.

The possibility to educate other young researches in CATFIS project will be investigated along the four-year period of CATFIS project.
2. Work plan

The general work plan of CATFIS project for 2015-2018 and a more detailed work plan of workpackages for year 2017 are presented below. The overall strategy in this project is to enhance nuclear safety by experimental and modelling studies. The experimental data produced in the workpackages will be utilized in the development of models or in severe accident analyses in general. All the proposed research topics of the project are covering phenomena which are poorly known. The costs of CATFIS project are described as required for each project year during 2015-2018 in Annex 3 (in Finnish). The planned funding is also described as required for each project year in Annex 3. The planned expenses and funding for each task in 2017 is given in an Excel sheet attached to this document as Annex 2-1.

2.1 Primary circuit chemistry of iodine (WP1)

The aim in this workpackage is to investigate the behaviour of iodine in the primary circuit conditions and validate/develop computer models based on the experimental data. After the Fukushima severe accident, the research priority classification was reviewed e.g. as part of SARNET community and the iodine chemistry was kept as a high priority issue. Especially, the need to study the effect of control rod material (B$_4$C) on the release of gaseous iodine has been emphasized. The effect of molybdenum on iodine release will be investigated a later phase of the project starting in 2017. Molybdenum behaviour is known to be highly dependent on the oxygen potential [Matzke, 1995]. It is expected that the presence of air would likely alter the reaction between molybdenum and caesium iodide in the primary circuit. These phenomena are not well-known internationally and thus the computer models to describe the experimental observations should be developed. The code development will be performed in collaboration with IRSN (starting in 2018).

Concerning iodine chemistry, the effect of boron on the release of gaseous iodine from CsI deposits on the circuit surface was examined in 2015-2016. A special interest was on the reaction temperature of caesium and boron and on the formation of solid Cs-B compound. Additionally, the presence of excess Cs to I to reduce the release of gaseous iodine was investigated.

Partners and person months allocated to WP1 for 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months (2017)</th>
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</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.8</td>
</tr>
</tbody>
</table>

2.1.1 Iodine chemistry (T1.1)

In 2017, the main goal is to study the effect of the initial Mo/Cs molar ratio on the release of gaseous iodine from a precursor mixture of caesium iodide and molybdenum oxide at 650 °C. The aim is to find out the effect of increasing molybdenum content (Mo/Cs ratio from 1.5 to 5) on the iodine behaviour. At the same time the experiments will clarify if further studies with different Mo/Cs ratios are needed. It is expected that the composition of atmosphere has an influence in the process on the formation of cesium molybdate in steam-rich atmosphere, as it has been mentioned in the literature. Consequently, the experiments in the air-containing atmosphere will provide relevant data for the comparison of results. It is expected that the higher Mo/Cs ratio, the higher amount of released gaseous iodine there will be. Likewise, the more steam-rich atmosphere, the higher amount of released...
gaseous iodine is expected to be formed. These assumptions will be verified and quantified with experiments. The experiments will be carried out with EXSI-PC facility. The measurement set-up is unique and it includes three identical sampling lines. The transported reaction products will be analysed both online and offline with aerosol and gas phase measurement devices (e.g. SMPS, ELPI, TEOM and FTIR) as well as with Raman, XRD, SEM-EDX and ICP-MS methods. A preliminary test matrix for EXSI-PC studies is presented in Table 2.1. The results will be presented in the international NUGENIA TA2.4 forum in order to enhance the interpretation of experimental results together with IRSN and JAEA, which are studying the corresponding gas phase reactions. As it was said above, VTT is focused on the reactions of fission product deposits taking place on the surface of primary circuit.

### Table 2.1. Preliminary test matrix of EXSI-PC studies.

<table>
<thead>
<tr>
<th>Exp</th>
<th>Precursor</th>
<th>T [°C]</th>
<th>Mo/Cs</th>
<th>Cs/I</th>
<th>Gas</th>
</tr>
</thead>
<tbody>
<tr>
<td>EXSI-1</td>
<td>CsI + MoO₃</td>
<td>650</td>
<td>5</td>
<td>1</td>
<td>Ar/Air</td>
</tr>
<tr>
<td>EXSI-2</td>
<td>CsI + MoO₃</td>
<td>650</td>
<td>5</td>
<td>1</td>
<td>Ar/H₂O</td>
</tr>
<tr>
<td>EXSI-3</td>
<td>CsI + MoO₃</td>
<td>650</td>
<td>1.5</td>
<td>1</td>
<td>Ar/H₂O</td>
</tr>
</tbody>
</table>

2.2 Formation of nitric acid (WP2)

The aim in this workpackage is to study the effect of beta radiation on the formation rate of nitric acid (HNO₃) in the containment building. The studies on nitric acid formation by radiolysis have previously been conducted with gamma radiation. However, MELCOR/ORIGEN calculations estimate that the dose rate in the containment atmosphere is mostly originated from beta radiation. The contribution of gamma radiation on the dose rate is roughly an order of magnitude lower. The formation rate of nitric acid will be studied by exposing humid air or N₂ atmosphere to beta radiation. In the next phase, mixtures of water pool and air or N₂ atmosphere will be exposed to irradiation. The radiation source can be located in the gas phase or in the liquid phase. Also temperature, water concentration, nitrogen/oxygen molar ratio and dose rate as well as radiation dose will be varied in the experiments. A mathematical model describing the formation of nitric acid will be derived. The HNO₃ formation rate results will be utilized in Chempool calculations on the containment pool pH in Task 3.1. These calculations will be focused on Finnish NPPs. Furthermore, the experiments will probably produce data on the formation of other air radiolysis products as well.

The first experiments on HNO₃ formation in humid air at moderate temperature were done in 2015. The experiments of year 2016 are currently ongoing. The results of year 2015 indicated that a radiation dose of four orders of magnitude lower by beta than gamma radiation generates a similar amount of HNO₃ molecules. This is a significant difference in the HNO₃ yield and it verifies the importance of this study. The dependence of HNO₃ formation on the beta radiation dose rate was also observed. On the basis of these observations, the accumulation and formation of HNO₃ in the containment pool due to beta radiation in the atmosphere, or directly in the pool, should be experimentally investigated in order to be able to give more reliable estimations e.g. on the iodine source term.

Partners and person months allocated to WP2 for 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>2.4</td>
</tr>
</tbody>
</table>

2.2.1 HNO₃ formation by beta radiation in gas phase and sump (T2.1)

The highest concentration of nitrogen (N₂) can be found in the containment atmosphere, but some amount of N₂ is also dissolved in the sump water. The goal is to study the formation of nitric acid (HNO₃) by beta radiation both in the containment atmosphere and containment sump conditions. In 2017 the formation rate of nitric acid in humid air or N₂ atmosphere and the subsequent accumulation of HNO₃ into the sump water will be studied experimentally. The irradiation of gaseous sample with beta radiation (P-32 source) will take place inside the BESSEL facility of VTT. A glass vessel with gas volume and water volume will be placed inside the BESSEL facility. The
concentration of HNO₃ generated in the gas phase will be measured e.g. online as a function of cumulative radiation dose using a FTIR. The subsequent accumulation of nitric acid into the water volume will be analysed after the experiments. The aim is to start the experiments at room temperature. The composition of the gas phase above the water volume will be varied between e.g. humid air and nitrogen. Also the impact of dose rate will be examined. The results will be compared with the previous data on HNO₃ formation in sump water by gamma radiation. If needed, it is also possible to repeat some of the beta radiation experiments with gamma radiation. As a result of this task, the formation rate of HNO₃ at various conditions will be defined. The gathered data will be used to develop a mathematical model for the observed formation rate of HNO₃. The model will also be useful in plant assessments on the pH of water pools. The obtained HNO₃ formation rate results will be used in Chempool calculations on pool pH as part of Task 3.1, in which the aim is to recalculate the previous severe accident scenarios of Lovisa and Olkiluoto NPPs.

2.3 Development of Chempool software (WP3)

The aim in this workpackage is to improve the chemistry model of ChemPool software [Penttilä et al., 2013] in the course of project (2015-2018). The work will include the improvement of the iodine chemistry model. Also the recent observations on Cs speciation when released from the primary circuit to the containment atmosphere will be considered. Nitric acid formation rates will be updated based on the experiments to be conducted with beta radiation in Workpackage 2. Another aim is to use the updated chemistry model of ChemPool to recalculate Olkiluoto and Lovisa severe accident scenarios.

The first ChemPool calculations utilizing the gained new experimental results and model in WP2 on HNO₃ formation in humid air by beta radiation at low temperature (<50°C) verified that the amount of NaOH needed to control the containment pool pH may be higher than expected.

Partners and person months allocated to WP3 for 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months (2017)</th>
</tr>
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<tbody>
<tr>
<td>VTT</td>
<td>0.9</td>
</tr>
</tbody>
</table>

2.3.1 Recalculation of Olkiluoto and Lovisa SA scenario (T3.1)

As a first goal in 2017, the nitric acid formation rate in ChemPool software will be updated. The formation rate will be received from the measurements on nitric acid formation by beta radiation in humid air and in water in WP2. The main emphasis in 2017 is on the evolution of pool pH when the atmosphere above the pool is irradiated. Currently, the HNO₃ formation rate in ChemPool is based on NUREG report [Beahm et al., 1992] that uses gamma radiation only. As mentioned above, MELCOR/ORIGEN calculations estimate that the dose rate in the containment atmosphere is produced mostly from beta radiation. Gamma dose rate is lower roughly with an order of magnitude. This could mean that the total formation rate of HNO₃ is higher than previously anticipated.

The second goal in 2017 is to recalculate Olkiluoto or Lovisa severe accident scenario with the updated HNO₃ formation rate by beta radiation in gas phase and pools. As a result, the calculations will give an overall view on the effect of various phenomena to pool pH and its subsequent effect on iodine speciation (or even on the iodine release from the pool) concerning especially Finnish NPPs. In addition, the measures to control the pool pH due to the calculation results will be assessed. The outcomes will be compared with the previous calculations.

2.4 Pool Scrubbing (WP4)

The aim in this workpackage is to investigate the retention of fission products due to the pool scrubbing phenomenon (e.g. containment pool and FCVS scrubber). There are some unknown features related to the pool scrubbing phenomenon [Jacquemain et al., 2016]. One of the internationally recognized open questions is the long-term behaviour of the pool. In the course of the accident evolution, the pool will get loaded with fission prod-
ucts and other materials, such as structural materials. The question is how these materials will effect on the trapping efficiency of the pool, is it possible that the pool will act more like a source of fission products. Another open question is the behaviour of organic iodine, which is difficult to trap. The behaviour of organic iodine in the pool should be investigated also considering the long-term operation of the pool. Additionally, the existing experiments have usually been performed at low temperatures, such as at 20 °C water temperature. The research should be extended to cover also realistic pool temperatures at boiling point or close to it. Also the effect of heat and radiation in oxygen rich or low pool conditions need to be examined to verify the possible decrease in the pool trapping efficiency for fission products. Radiation and heat together with oxygen may change the alkaline chemistry of the pool. After mapping the conditions where the problem in the pool trapping efficiency will arise, the possible means to solve the problem will be examined. Another aim in this workpackage is to perform experiments considering the special features of Finnish NPPs, such as pool chemistry.

Partners and person months allocated to WP4 for 2017 are given in the table.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months (2017)</th>
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<tbody>
<tr>
<td>VTT</td>
<td>2.1</td>
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</tbody>
</table>

2.4.1 Iodine (gas/aerosol) retention in Containment Pool and FCVS (T4.1)

The goal in 2017 is to focus on the pool scrubbing phenomenon with a slight emphasis on the FCVS scrubber. However, the experimental data will cover several conditions, thus it will be applicable for the containment pool analyses as well. Three experimental series will be performed partially, see the preliminary experimental matrix below (Table 2.4). The importance of parameters will be seen during the experiments and thus, the experiments to be performed will be clarified in the course of project. The aim is not to do all the combinations of experiments shown in Table 2.4. The main difference between the series is in the initial pool pH, which will be either pH 7 or 13, in order to point out the difference in the trapping efficiency due to pH. Depending on the amount of performed experiments, the possibility to perform some comparison experiments at acidic conditions will also be explored. The desired alkaline pool pH will be achieved by mixing a sodium hydroxide (0.5 %) and sodium thiosulfate (3.5 %) solution. Another important parameter is temperature; it will be varied from 20 to 100 °C (boiling at 1 bar absolute pressure). The pressure will be kept close to atmospheric pressure. The retention of CsI aerosol and gaseous CH$_3$I organic iodine in the pool will be studied. The experiments will be continued as long as the release of the above mentioned fission product representatives from the pool is significant when compared with the feed of them into the pool.

One experimental series focuses on to study the effect of oxygen content on the alkaline chemistry of the pool when exposed to heat and gamma radiation. The aim is to find out if the trapping efficiency of the pool decreases. In the experiments the pool will be flushed either with nitrogen or air and at the same time heated and irradiated before the retention of FPs will be investigated. The pool solution will be analysed at this stage and it will give information on the possible changes in pH from pH 13 etc. Next, the same pool will be injected with CsI or CH$_3$I when exposed to heat and irradiation again and thus the possible decrease in the trapping efficiency of the pool when compared with the other experimental series (see above) will be observed.

Other possible parameters to be varied in the experiments are flow rate through the pool, the height of the water column in the pool, the diameter of gas feeding line into the pool and its ratio to the pool size. These parameters are of secondary importance and probably they will not be varied in all experiments. However, the parameters are important on the pool scrubbing phenomenon in general. For example, they have a significant impact on the pool hydrodynamics (gas bubble size, bubble residence time in the pool, etc.), but it is not possible to consider all the parameters in detail within one year. Therefore, VTT will start collaboration about knowledge transfer with PSI, which will offer technical assistance to VTT in hydrodynamics. PSI has studied the pool hydrodynamics for several years. In order to strengthen the collaboration with PSI, VTT will apply NUGENIA researcher mobility grant when the call will be opened in 2017 and the aim is to organize a researcher visit at PSI with duration of 2 to 3 months. This possibility has been discussed and agreed with PSI.
The gathered results will be presented also in the NUGENIA TA2.4 area IPRESCA project (starting in 2017) meetings. The IPRESCA project is dedicated to summarize the current knowledge on pool scrubbing phenomenon internationally and to clarify the future research needs. Currently no external funding is foreseen in the IPRESCA project. Partners are expected to join the project with in-kind contributions. Nevertheless, based on the progress made within IPRESCA, partners may seek international agencies and organizations to fund the common partnership project in the future.

These experiments will also provide data to the SAFIR2018 RG2 CASA project, in which the WP4 experiments will be simulated with the MELCOR and ASTEC codes in 2017. The goal is to verify the performance of the codes when simulating pool scrubbing and also to compare the results of the codes. The experimental and modelling results will also be shared through OECD/NEA THAI-3 program, which is followed-up by the CASA project.

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<th>boatling</th>
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</tr>
<tr>
<td>pH, neutral</td>
<td>7 7 7 7</td>
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<tr>
<td>T [°C]</td>
<td>20 50 80 100</td>
<td></td>
</tr>
<tr>
<td>Air [l/min]</td>
<td>1,4,8,16 1,4,8,16 1,4,8,16 1,4,8,16</td>
<td></td>
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<tr>
<td>Water column [cm]</td>
<td>10,20,40 10,20,40 10,20,40 10,20,40</td>
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Separate experiments

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<tbody>
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<td>CsI CsI CsI CsI</td>
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<tr>
<td>Gas</td>
<td>CH3I CH3I CH3I CH3I</td>
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<tbody>
<tr>
<td>Duration</td>
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<tr>
<td>Column to orifice diameter ratio</td>
<td>5,7 5,7 5,7 5,7</td>
<td></td>
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<tr>
<td>pH, alkaline</td>
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<tr>
<td>T [°C]</td>
<td>20 50 80 100</td>
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</tr>
<tr>
<td>Air [l/min]</td>
<td>1,4,8,16 1,4,8,16 1,4,8,16 1,4,8,16</td>
<td></td>
</tr>
<tr>
<td>Water column [cm]</td>
<td>10,20,40 10,20,40 10,20,40 10,20,40</td>
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Separate experiments

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<tbody>
<tr>
<td>Aerosol</td>
<td>CsI CsI CsI CsI</td>
<td></td>
</tr>
<tr>
<td>Gas</td>
<td>CH3I CH3I CH3I CH3I</td>
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<tr>
<td>Duration</td>
<td>as long as the release is 50% of the feed</td>
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<tr>
<td>Column to orifice diameter ratio</td>
<td>5,7 5,7 5,7 5,7</td>
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Table 2.4. The preliminary experimental matrix of pool scrubbing experiments.

2.5 OECD/NEA STEM-2 and BIP-3 follow-up (WP5)

The workpackage aims at following the progress of the four-year OECD/NEA STEM-2 experimental program, which is a continuation of STEM program [Clément and Simondi-Teisseire, 2010]. STEM-2 program started in January 2016. Another aim is to follow-up the progress of the three-year OECD/NEA BIP-3 experimental program, which is a continuation of BIP and BIP-2 programs. BIP-3 program started in January 2016. VTT will represent Finland in the follow-up meetings of both programs. The progress of the programs will be reported in SAFIR2018 program. The specific project plans of both programmes were distributed to SAFIR2018 RG2 members in summer 2015.
The travel accounts of the first STEM-2 and BIP-3 meetings in 2016 have been distributed to SAFIR2018 RG2. The main message in both programmes was that the experimental work is starting and the participants were discussing about the experimental matrixes to be performed in the programmes. The experimental work is ongoing now and the results will be delivered starting in 2017.

Partners and person months allocated to WP5 for 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP5</th>
<th>Person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.0</td>
</tr>
</tbody>
</table>

2.5.1 Participation in the STEM-2 meetings (T5.1)

VTT will participate in OECD/NEA Source Term Evaluation and Mitigation 2 (STEM-2) interpretation circle meetings. In these meetings, VTT will also present results from the iodine and ruthenium studies carried out in CHEMPC, TRAFI and CATFIS projects. These results will complete the database of STEM-2 program. The participation fee of Finland for year 2017 is 6200 euros. The fee as a whole (100%) will be charged from VYR.

SAFIR RG2 has seen STEM-2 program as an important topic to follow and CATFIS project is willing to perform the follow-up. However, the participation fee should not decrease the amount of funding allocated by VYR for the experimental work of CATFIS in other workpackages.

2.5.2 Participation in the BIP-3 meetings (T5.2)

VTT will participate in OECD/NEA Behaviour of Iodine Project 3 (BIP-3) interpretation circle meetings. In these meetings, VTT will also present results from the iodine studies carried out in CHEMPC, TRAFI and CATFIS projects. These results will complete the database of BIP-3 program. The participation fee of Finland for year 2017 is 3400 euros. The fee as a whole (100%) will be charged from VYR.

SAFIR RG2 has seen BIP-3 program as an important topic to follow and CATFIS project is willing to perform the follow-up. However, the participation fee should not decrease the amount of funding allocated by VYR for the experimental work of CATFIS in other workpackages.
3. Deliverables and milestones 2017

The proposed list of deliverables for 2017 is presented in Table 3.1. The results of the performed work in CATFIS project will be summarized in four reports. The main findings will also be presented in at least one scientific publication. Travel accounts on the OECD/NEA STEM-2 and BIP-3 follow-up meetings will be prepared.

Table 3.1. List of deliverables in CATFIS project during 2017.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Progress report on primary circuit chemistry of iodine – Milestone: The report will be completed by 15.9.2017</td>
<td>1.3</td>
<td>15.9.2017</td>
</tr>
<tr>
<td>D1.1.2</td>
<td>Publication on primary circuit chemistry of iodine</td>
<td>0.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Progress report on formation of nitric acid by beta radiation</td>
<td>2.4</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Progress report on development of Chempool code and on calculations of severe accident scenarios</td>
<td>0.9</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D4.1.1</td>
<td>Progress report on pool scrubbing</td>
<td>2.1</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the facility is built and the first set of the total set of experiments on pool scrubbing in WP4 is done by the end of April 2017</td>
<td>N/A</td>
<td>30.4.2017</td>
</tr>
<tr>
<td>D5.1.1</td>
<td>Travel account of OECD/NEA STEM-2 meeting – The minutes of the meeting will be distributed to RG2 by 30.9.2017 (selected as milestone)</td>
<td>0.5</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D5.2.1</td>
<td>Travel account of OECD/NEA BIP-3 meeting – The minutes of the meeting will be distributed to RG2 by 30.9.2017 (selected as milestone)</td>
<td>0.5</td>
<td>30.9.2017</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Total pm</td>
<td></td>
<td>8.0</td>
</tr>
</tbody>
</table>
4. Project organisation

The project manager is a PhD student (MSc Tech) Teemu Kärkelä from VTT Technical Research Centre of Finland. The deputy project manager is MSc Tommi Kekki (VTT). The organisation responsible for the whole project is VTT.

Partners in joint activities are:
1) The primary circuit chemistry of iodine will be studied in collaboration with JAEA (Japan) and IRSN (France). The collaboration will focus on the interpretation of results and on the development of experimental techniques at first. The action preferably takes place through the international NUGENIA TA2.4 forum. Starting from 2018 the experimental data will be utilized to develop mathematical models together. VTT aims at to get the models implemented to the ASTEC/SOPHAEROS module. IRSN is one of the main developers of ASTEC code, thus a direct collaboration with IRSN is expected.

2) In WP4 VTT will start experiments on the pool scrubbing phenomenon. Since the understanding of the phenomenon needs also knowledge of the pool hydrodynamics, VTT will be in collaboration with PSI (Switzerland) on the interpretation of results and thus VTT’s fission product behaviour in pool data can be combined with PSI’s hydrodynamics knowledge. The collaboration will take place as knowledge transfer between the organizations, later in 2017 a NUGENIA researcher mobility at PSI will be applied. The VTT experimental data of WP4 will be utilized in the SAFIR2018 RG2 CASA project, in which the experiments will be simulated with MELCOR and ASTEC codes and the differences between the code calculation results will be compared. The results of pool scrubbing study in WP4 will be shared in the NUGENIA TA2.4 IPRESCA project, which is an international forum dealing with the pool scrubbing phenomenon mapping the current knowledge and the areas of lacking information. The outcomes of the project will be reported to the SAFIR2018 program. Currently no external funding is foreseen in the IPRESCA project. Partners are expected to join the project with in-kind contributions. The experimental and modelling results of WP4 will be also shared through the OECD/NEA THAI-3 program by the CASA project.

The main researchers of the CATFIS project and their title, organisation and tasks are listed in Table 4.1. The estimated person months of the project personnel for year 2017 are presented as well.

Table 4.1. The personnel of CATFIS project during 2017.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Teemu Kärkelä</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>2.1, 4.1, 5.1</td>
<td>2.7</td>
</tr>
<tr>
<td>Tommi Kekki</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>2.1</td>
<td>0.8</td>
</tr>
<tr>
<td>Melany Gouello</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>1.1, 5.2</td>
<td>1.0</td>
</tr>
<tr>
<td>Jouni Hokkinen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>1.1, 4.1</td>
<td>2.6</td>
</tr>
<tr>
<td>Karri Penttilä</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>3.1</td>
<td>0.9</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>8.0</strong></td>
</tr>
</tbody>
</table>
5. Risk management

The personnel resources for the work proposed are sufficient. The new workpackage 4 “Pool Scrubbing” has gained person-months from WP1. The research personnel compose of MSc and PhD researchers with a long experience on source term studies. Most of the personnel have participated in the two previous SAFIR programs at least. The experimental facilities in workpackages 1 and 2 exist already, in case of the new WP4 the main parts of the facility exist and it will be straightforward to set up the facility. Thus the risks for the experimental work are low. For the modelling part of the work e.g. ChemPool, all computer programs needed are already available. The progress of CATFIS project will be followed in SAFIR2018 reference group meetings. In addition, VTT’s internal follow-up meetings of CATFIS project will be held monthly.
References


Bosland, L., Cantrel, L., Girault, N., Evaluation of the dose rate inhomogeneities in Phébus containment during FPT-1 and FPT-3 tests, Presentation in the 31th CCIC meeting (2010), restricted distribution.


### Project Name

**Work packages and Tasks**

<table>
<thead>
<tr>
<th>WP1 - Primary circuit chemistry of iodine</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat&amp;supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Memb fee</th>
<th>Other</th>
<th>TOTAL</th>
<th>VYR</th>
<th>Ferrovial</th>
<th>Fortum</th>
<th>TVO</th>
<th>Aalto</th>
<th>LUT</th>
<th>VTT</th>
<th>NKS</th>
<th>Other</th>
</tr>
</thead>
<tbody>
<tr>
<td>T1.1 Iodine chemistry</td>
<td>1.8</td>
<td>23.0</td>
<td>2</td>
<td>2</td>
<td></td>
<td></td>
<td>3.2</td>
<td>30.2</td>
<td>20.2</td>
<td>10.0</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

| WP2 - Formation of nitric acid           | 2.4    | 31.0      | 3        | 0      |          |          | 4.4    | 38.4  | 25.4| 13.0      |
| T2.1 HNO₃ formation by beta radiation in gas phase and sump | 2.4    | 31.0      | 3        |        |          |          | 4.4    | 38.4  | 25.4| 13.0      |

| WP3 - Development of Chempool software   | 0.9    | 11.5      | 0        | 0      |          |          | 1.6    | 13.1  | 7.1 | 6.0       |
| T3.1 Recalculation of Olkiluoto and Loviisa SA scenarios | 0.9    | 11.5      |          |        |          |          | 1.6    | 13.1  | 7.1 | 6.0       |

| WP4 - Pool Scrubbing                     | 2.1    | 26.8      | 2        | 1      |          |          | 3.8    | 33.6  | 23.6| 10.0      |
| T4.1 Iodine (gas/aer) retention in Containment Pool and FCVS | 2.1    | 26.8      | 2        | 1      |          |          | 3.8    | 33.6  | 23.6| 10.0      |

| WP5 - OECD/NEA STEM-2 and BIP-3 follow-up| 1.0    | 12.4      | 0        | 4      |          |          | 9.6    | 17.7  | 23.7| 4.0       |
| T5.1 Participation in the STEM-2 meetings | 0.5    | 6.2       | 2        |        |          |          | 6.2    | 0.9   | 15.2| 13.3      |
| T5.2 Participation in the BIP-3 meetings  | 0.5    | 6.2       | 2        |        |          |          | 3.4    | 9.6   | 12.4| 10.5      |

| TOTAL                                  | 8.0    | 104.6     | 7.0      | 7.0    | 0.0      |          | 9.6    | 148.0 | 143.0| 100.0     |

**Comments:**

Memb fee: The participation fees of Finland for OECD STEM-2 and BIP-3 programmes in 2017 are 6200 eur and 3400 eur, respectively. The STEM-2 and BIP-3 fees as a whole (100%) will be charged from VYR.

WP1 and WP4 Travel costs are due to presentation of iodine and pool scrubbing results in the coming international conferences/meetings: NENE 2017, ICAPP 2017, NUGENIA TA2.4 meetings.
SAFIR2018 Project plan

COVA

Comprehensive and systematic validation of independent safety analysis tools

Seppo Hillberg, Joona Kurki, Ismo Karppinen, Ari Silde

VTT Technical Research Centre of Finland
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1. Research theme and motivation

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 important 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 Fennovoima, 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.

1.1 Background and state-of-the-art

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

A very large number of validation experiments including both separate effect tests (SET) and tests performed in integral test facilities (ITF) have been calculated during the long development history of the code. The problem with these validation cases is that each of them has been calculated with the version of Apros that was the most recent at the time, and thus somewhat different results may be expected with newer version of the code. Furthermore, the validation of the most fundamental physical models in the code has been done in the very first years of the code’s development, or in some cases simply bypassed with the justification that the same models have been successfully utilized in other well-validated codes.

More systematic approach with proper quantification of the prediction error would result in better overall understanding on the limits of the code. The process will also provide basic understanding of the essential physical phenomena in the separate effect tests and integral test facilities. Furthermore, quantification of the prediction error in the case of the most basic separate effect tests would result in direct estimation of the input uncertainty distributions of the physical models in the code. These uncertainty distributions could then be utilized as a-priori knowledge when trying to quantify the input uncertainties of other physical models in more complex situations where multiple physical models have simultaneous effect on the simulation results, and also as direct input for Best Estimate Plus Uncertainty (BEPU) analyses.

TRACE\(^1\) (TRAC/RELAP advanced computational engine) is a system-scale safety analysis tool developed by United States Nuclear Regulatory Commission (U.S. NRC), largely similar to Apros but limited in its intended scope of use. It is a successor of TRAC and RELAP5 code families, and has been extensively validated by its developers for safety analyses of light water reactors of the types currently in operation in the United States. Because of the massive number of man-years that U.S. NRC has been able to put into TRACE’s verification and

\(^1\) TRACE is accompanied by a neutron diffusion code called PARCS that provides the prediction of the neutronic behaviour of the reactor core. Even when not explicitly stated, the name TRACE is used to refer to the coupled code TRACE/PARCS in this document.
validation, and the rigorous approach the code’s developers have put into this work, TRACE can be considered as representing the international state-of-the-art of a well-validated system code for its intended scope of use.

Code-to-code comparisons of experiments calculated with both Apros and TRACE would help to ensure that capabilities of Apros are on par with the state of the art, and possibly to reveal aspects in the code that might call for improvement. Analyses of nuclear power plants or test facilities representing nuclear power plants that are outside the main intended scope of TRACE’s use, such as for example VVER-type reactors, would be beneficial in forming an insight how well the code is able to model these kinds of reactor systems.

U.S. NRC’s CAMP (Code Applications and Maintenance Program) programme provides the participating countries with access to NRC’s safety analysis tools such as TRACE, RELAPS, PARCS and their graphical user interface SNAP. The current CAMP agreement period ends in September 2017. Negotiations for agreement period 2017-2022 are expected to begin around turn of the year 2016-2017. Participation in the program is included in this project proposal.

OECD’s Nuclear Energy Agency (NEA) organizes nuclear safety research through multiple experimental and theoretical research projects. In the recent years, activities organized by NEA and its subcommittees relevant to thermal hydraulics of nuclear reactor safety include the ATLAS (Advanced Thermal-hydraulic test Loop for Accident Simulation), BEMUSE (Best Estimate Methods – Uncertainty and Sensitivity Evaluation), HYMERES (Hydrogen Mitigation Experiments for Reactor Safety), PKL (Primärkreislauf-Versuchsanlage), PREMIUM (Post BEMUSE Reflood Models Input Uncertainty Methods) and UAM (Uncertainty Analysis in Modelling) projects, and Working Group on Analysis and Management of Accidents (WGAMA). Participation in HYMERES Phase 2 programme is included in the project proposal.

FONESYS (Forum & Network of System Thermal-Hydraulics Codes in Nuclear Reactor Thermal-Hydraulics) is an international network of code developers with an aim to highlight the capabilities and the robustness as well as the limitations of current system-scale thermal-hydraulic codes to predict the main phenomena during transient scenarios in nuclear reactors for safety issues. VTT has participated in FONESYS since its formation in 2010 as developer of the Apros simulation code. Continued participation in the network is considered essential for keeping in touch with developers of other similar simulation tools, to share knowledge and experiences on code development and use, and in this way furthering nuclear safety on its part. FONESYS also organizes benchmarks to evaluate code capabilities in modelling challenging numerical and physical phenomena.

1.2 Objectives and expected results

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

The project is commenced by carrying out a critical assessment of the state of Apros’ validation. A roadmap for further validation work is devised with the help of the system code validation matrices published by OECD/NEA, with ranking potential validation cases based on the importance of experiment from the point of view of overall validation, and also taking into account accessibility of the experimental data. While the main body of the assessment work was finished in the first year, the roadmap will be refined in the following years as the work goes on.

The work results in critical review of Apros’ state of validation, with a suggested roadmap on what experiments could and should be calculated for furthering the validation for relevant nuclear power plant applications. Work on implementing the roadmap is done in a systematic and rigorous way that results in quantification of the output uncertainties of the simulation results.

All the analysed validation cases are thoroughly documented, including estimates of the output uncertainties and any aspects in the code or in the experiment that should be taken into account when interpreting the results of the validation simulations and their significance. These validation documents then form a database that can be used as a reference when applying the code to real-life analysis purposes. The validation process involves training of new experts in the thermal hydraulic area with good understanding of the basic phenomena and capable of applying the knowledge widely instead of producing mere well-trained users of a single code.

As it is of utmost importance that the user effect and risk of analysis errors is minimized when carrying validation analyses that aim to serve a greater purpose as reference material for all current and future users of system codes, special attention is placed on this aspect; international recommendations for user effect reduction and
guides on how deterministic safety analyses should be carried out are kept as guidelines for the successful conduction of the work.

For situations in which the uncertainty in the simulation results in a validation analysis can be attributed to a single physical model within the code, that model’s input uncertainty distribution is determined. For more complex situations the input uncertainties have to be determined using methods based on statistical techniques taking into account a-priori information on those physical models that have already been determined. For such case, the plan is to use methods developed in or suggested by the SAFIR2018 project USVA (Uncertainty and Sensitivity Analyses for reactor safety).

New thermal-hydraulic experiments will be carried out during the project lifetime in Finland in SAFIR2018 and abroad, often as part of international research projects, in order to investigate a particular reactor type’s or component’s behaviour in a particular accident, transient or steady-state scenario, or to take into account factors that have been neglected in earlier experiments, such as non-condensable gases. Simulation of such experiments can be used to extend the validation range of analysis codes to new applications. Furthermore, analysing new experiments both in pre-test phase as “blind” calculation, and in post-test phase provides invaluable insight into what are the capabilities of both the simulation code and the person carrying out the analysis; reasons for discrepancies between the blind calculation predictions and the experimental results should be deduced when carrying out the post-test analysis, revealing whether the cause is in the inherent limits of the code’s capabilities, in the experience of the modeller, or perhaps in the conduction of the experiment itself. Finally analysing a new experiment with two different codes enables direct comparison between the codes’ performance, which can possibly reveal shortcoming in either or in both of the codes.

The new experiments that can be analysed in COVA are largely mandated by what experiments are carried out in other projects in SAFIR2018 and in the international research projects in which VTT participates, and thus has access to data. Still when choosing which of the possible experiments are to be analysed, importance is put into i) in what extent the experiment has relevance to the power plants in use or planned in Finland, ii) whether the experiment includes use of passive safety systems and iii) whether the experiment includes study of non-condensable gas effects.

1.3 Exploitation of the results

The primary end users of this work are domestic safety authorities, power plant operators and research organizations. The analysis codes validated in this project can be used for example in safety analyses carried out for operating nuclear power plants and in the assessment of construction and operating licence applications, and for various nuclear-safety related research purposes.

The validation work results in better understanding of analysis codes capabilities and limitations by the entire domestic thermal hydraulic expert community, and in better understanding of the basic physical phenomena by the new experts involved in the project. The results will be documented in a systematic way that helps also other users of the analysis codes in carrying out similar analyses. Identification of lacking or limited features in the codes can be used to guide further core development efforts. The Apros models of validation cases and the obtained results will be saved in a database and put available for all Apros users.

For Apros, the work results also in quantification of input uncertainties of some of the physical models within the code. This information is also documented and can be utilized as input data in best estimate plus uncertainty analyses as well as in further work that aims in quantifying the input uncertainties of Apros’ physical models.

All results produced in the project are applicable to real-life problems by all end users.

1.4 Appropriateness of the project to SAFIR2018 programme

The COVA projects fits perfectly in the SAFIR2018 programme, as the work carried out in this project is directly aimed at helping to ensure the availability of well-validated tools and skilled experts that can utilize them in analyses of any emerging safety issues related nuclear power plants. The project fosters several new experts to this area currently affected by the retirement of the previous generation of experts.

In particular, this project answers in its own part in multiple research goals and aspects highlighted within the SAFIR2018 framework plan:

- Validation of independent thermal-hydraulic calculation codes, and especially modelling of new types of safety systems has been deemed as still requiring more resources
- Validation of all software used in computational nuclear safety analyses is encouraged to be developed to a more systematic and extensive direction
• Use of OECD/NEA's thermal-hydraulic validation matrices, the content of which have been last time updated in 2010, is encouraged
• Participation in computational analyses of reactor safety experiments carried out in international research projects is encouraged to get full benefit from these projects
• The project promotes international networking and cooperation through participation in international research projects such as the CAMP programme and projects coordinated by OECD NEA. Experimental data produced in these projects is also essential input for the work performed in this project
• The project supports, in its own part, experimental activities carried out at Lappeenranta University of Technology. Also experiments carried out at LUT provide useful input for this project.
• The project has an important educational aspect that aims at competence transfer from an older generation of scientists to younger researchers (see Section 1.5)

1.5 Education of experts

This project has an essential educational aspect through learning-by-doing; the key tasks carried out by younger researchers in this project help them to form a fundamental understanding of a thermal-hydraulic systems scale analysis code's capabilities and limitations from the nuclear safety point of view. Such know-how is practically impossible to develop otherwise. Also it should be noted that the kind of work being done in this project was last time carried out some 20 to 30 years ago, and the researchers involved in those endeavours are nearing their retirement age, calling for an urgent need for competence transfer in this field.

The topic of this project is such that M.Sc. theses can be written based on the work and scientific research articles can contribute to dissertations of people involved in the project. Realization of theses depends highly on the responsible organization's ability to hire new trainees and PhD students during the project years.

In addition to M.Sc. theses, the results of this work are to be disseminated through scientific journal and conference articles. The work with comprehensive validation of Apros is expected to result in at least two scientific publications towards the end of the project. Publications are foreseen related to work on new experiments as well as international cooperation projects. A summary article on the project will be published in a journal at the end of the fourth year of the project.
2. Work plan

COVA is divided into four work packages (WP’s): Validation matrices (WP1), Analyses of new experiments (WP2), Management and international cooperation (WP3) and Participation fees (WP4). The actual research work dealing with analysis tool validation is carried out in the first two work packages, with WP1 concentrating on the fundamental aspects of the validation work with Apros, and WP2 in application of Apros and TRACE to validation using primarily integral-scale experiments with proper quantification of output uncertainties. Work package 3 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. Work package 4 includes the participation fees of international research projects and nothing else.

Experience and direct results, such as the quantified uncertainty distributions of the input parameters, from the fundamental validation work done in WP1 can be utilized in best-estimate plus uncertainty analyses in WP2. Work performed in WP2 supports the international activities of WP3, and vice versa: experimental data obtained through the international projects in WP3 is utilized as input in WP2. The participation fees funded from WP4 make it possible to participate in the international projects in WP3. All activities in work packages 1 through 3 are essential in advancing the project objectives outlined in Section 1.2: improving validation of Apros and TRACE and promoting a more systematic and rigorous approach to the validation work.

2.1 Validation matrices (WP1)

Work package 1 aims at comprehensive and systematic validation of Apros. By the end of the first year, roadmaps for validation of Apros’ thermal-hydraulic and containment models have been crafted by critically assessing the current state of validation of these models, and reflecting these to the validation matrices published by OECD/NEA [1-6]. These roadmaps are then implemented during the four year project by calculating as many as possible of the chosen validation cases in order of importance, starting in the first year from the separate effect tests dealing with basic phenomena, and gradually working toward more complex experiments.

The extent of WP1 in terms of person months is listed in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>10.5</td>
</tr>
</tbody>
</table>

2.1.1 Critical assessment of TH model’s validation (T1.1)

The goal of task T1.1 is to critically assess the state of validation of Apros’ thermal-hydraulic model, and reflect it to the thermal-hydraulic validation matrices published by OECD/NEA.

Main body of the work was was done in 2015 and is reported in “Assessment of Apros thermal hydraulic models with separate effect tests” [7]. A small effort is reserved for updating the code validation database and the roadmap. When the validation work progresses in the task T1.3, it is anticipated that the experimental data for some of the chosen experiments may not be available.
2.1.2 Critical assessment of containment model’s validation (T1.2)

In task T1.2 the same approach that was used in task T1.1 for the thermal-hydraulic model, was repeated in 2015 for the Apros containment model. All validation work conducted so far was systematically described and assessed. The assessing work was conducted by means of the OECD/CSNI Containment Code Validation Matrix (CCVM) [6]. As a result, suggestions and recommendations for further validation were given. These suggestions include both the new experiments to be calculated and the older experiments, which are already calculated, but which should be re-analysed using the newest code version. This work is reported in “Assessment of Apros containment software” [8].

The assessment work is related only for those models, which are developed to be used in analysing purposes. The work does not cover some severe accident models, such as thermal effect of diffusion flame and indication of possibility to deflagration-to-detonation transition, which are very simple models and recommended only for training and simulator purposes. The main focus is in light water reactor phenomena in BWR and PWR containments.

Like in task T1.1 the work done in this task will be completed during the years 2016-2018 by adding new cases to the database and by searching relevant experiments to fill the gaps in code validation.

2.1.3 Validation analyses of TH model (T1.3)

2017 budget was reduced 20 k€ from the applied one and tasks T2.2 and T3.7 have been left out of the budget. The resulting 4.5 k€ has been moved to this task. The intention, if a suitable person is found, is to allocate whole T1.3 to the use of a diploma worker. Allocated person months of the task have been increased to 8. If no suitable person can be found for the position, realistic person month estimate is 4.5.

The roadmap defined in task T1.1 is implemented in task T1.3 during the four year project. Based on the evaluation done in the first year [7] it seems that the validation of some phenomena is based on quite few validation cases. Some of the phenomena are geometry-dependent and therefore need validation with different experiments. New validation cases are selected first from the basic phenomena, which do not have comprehensive validation. Publicity of the data is taken into account when selecting the experiments to be calculated. If feasible, in some suitable cases U.S. NRC’s TRACE code is used for code-to-code comparison. Using TRACE also enables creation of in-kind contributions which are required for the CAMP agreement. In year 2017 the goal is to hire a trainee to begin writing a Masters thesis on the subject of Apros’ validation based on separate effect tests. Hiring a trainee, however, depends on the organization’s ability to hire new employees. The thesis would be finished in 2018.

In year 2016 work in this task focused on core reflooding phenomena by calculating experiments made in ACHILLES, FLECHT-SEASET and ERSEC facilities. In year 2017 the focus stays in the related phenomena. Interfacial friction and water droplet entrainment are important phenomena defining flow conditions and affecting prediction of heat transfer during reflooding. Separate effect test recommended for code validation in phase separation in the core [7] are among others NEPTUN emergency core cooling heat transfer tests and UKAEA-Winfrith reflooding experiments. Winfrith post dryout experiments were conducted in a 9.75 mm tube. Void fraction was measured in three elevations with gamma densitometer making experiments suitable for evaluating interfacial friction model in addition to validating heat transfer. The NEPTUN reflooding experiments were performed in a half length 37 rod bundle. Data of both experiments is available in NEA databank and both tests have been used in validation of RELAP5 or TRACE, facilitating comparison to an other equivalent model and set of correlations. Modelling of selected experiments or other similar tests extends Apros validation in modelling reflooding phase of LOCA and serves as preparation for modelling PKL-4 i1 reflooding experiment in integral test facility (see 2.2.1)

Because the validation analyses are being mainly performed by younger researchers, an extra effort is put into ensuring that the simulation models are properly defined and calculations correctly performed, and so to minimize the risk of user error. To this end practices suggested in the OECD/NEA report “Good practices for user effect reduction” [9] are put in use in the project in the largest meaningful extent, and in addition all researchers joining the project familiarize themselves with the IAEA Specific Safety Guide on “Deterministic Safety Analysis for Nuclear Power Plants’’. Experience on code use gained during the project is documented in a project’s internal wiki page, with the aim that this would form a basis for a “best practices guideline” document that would be produce in the fourth year of the project.
2.1.4 Validation analyses of containment model (T1.4)

The roadmap defined in task T1.2 is implemented in task T1.4 by using the same methods as in task T1.3.

Because the validation analyses are being mainly performed by younger researchers, an extra effort is put into ensuring that the simulation models are properly defined and calculations correctly performed, and so to minimize the risk of user error. To this end practices suggested in the OECD/NEA report “Good practices for user effect reduction” [9] are put in use in the project in the largest meaningful extent, and in addition all researchers joining the project familiarize themselves with the IAEA Specific Safety Guide on “Deterministic Safety Analysis for Nuclear Power Plants” [10].

In year 2017, work in this task will focus on validation of sump heat and mass transfer calculation of the Apros containment model. Apros has a specific pool stratification model and the pool surface temperature is calculated iteratively due to its strong influence on the combined heat and mass transfer processes. Principles of this approach deviate from many common known lumped parameter containment codes. There has been earlier a lack of experimental data needed to validate the pool surface temperature, and further, pool condensation/evaporation calculation of Apros. However, IRSN has conducted a series of sump tests at the TOSQAN facility, where the wall condensation and sump evaporation have been investigated. Summary of these tests and calculation work has been now published as scientific journals, which give limited but necessary information on the boundary conditions and some test results. A selected TOSQAN sump test will be calculated by Apros containment model in year 2017.

2.2 Analyses of new experiments (WP2)

In work package 2, new experiments and benchmark exercises are analysed with the system codes Apros and TRACE. While the choice of analysed experiments is largely mandated by what experiments are carried out in the experimental projects in which VTT participates, and thus has access to data, preference in selection is done in favour of experiments dealing with functionality of passive safety systems and experiments in which non-condensable gases have a significant effect. Also, high priority is assigned to analysing experiments carried out at Lappeenranta University of Technology, in order to support domestic experimental activities. In 2017 and 2018 these activities include planned PWR depressurization experiment in SAFIR2018 INTEGRA project and containment spray experiments in SAFIR2018 INSTAB project. Possible benchmark exercises analysed in WP2 are, among others, those arranged by OECD/NEA, the FONESYS network and Atomic Energy Research (AER) project.

Best estimate plus uncertainty (BEPU) approach is used for estimation of uncertainties of the simulation results in analyses of new experiments in the extent that is practically feasible and meaningful: because of lack of information on the input uncertainties required in the BEPU analyses, repeating the computationally costly BEPU analyses may not be worthwhile in all possible analyses. Effort is put into simulating new experiments already in the pre-test phase (i.e. as “blind” calculation, before the actual experiment has been performed). This is helpful in forming a realistic understanding of the capabilities and limitations of both the tool used in the analysis and the person carrying out the simulation. Subsequent post-test analyses should reveal the reason for possible discrepancy between the pre-test predictions and experimental measurements, whether it is in the analysis tool, in the skills and experience of the modeller performing the analysis, in the practical implementation of the experiment, or in the specification of the experiment supplied in the pre-test phase.

Partners and person months allocated to WP2 are listed in the table below.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>5.0</td>
</tr>
</tbody>
</table>
2.2.1 PKL-4 (T2.1)

The first PKL-4 experiment, i1.1 run 1, focuses on the quench front propagation in the core during flooding in the course of a LB/IB-LOCA. GRS analyses of PERICLES experiments have indicated overestimation of water entrainment in the steam flow during core quenching. This is of relevance to the flooding and cooling of the core and therefore provides motivation for the study of the behaviour in the PKL-4 programme. The proposed experiment shall provide the basis for the evaluation of calculation models employed in the simulation of the quench-front propagation in the core and the water entrainment to the SGs [11]. The experiment i1.1 will be modelled with Apros. In addition it is foreseen that at least GRS will model the test and good cooperation in analysis and code to code comparisons are expected.

2.2.2 Support of LUT PWR depressurization experiment (T2.2)

As the corresponding task was left out of SAFIR2018 INTEGRA, this task will not actualize in 2017.

2.2.3 HYMERES (T2.3)

HYMERES programme studies complex safety relevant issues related to hydrogen in nuclear power plant containment with well instrumented large scale test facilities. In years 2015 and 2016 experiments made in MISTRA and PANDA facilities have been calculated in the COVA project. The current HYMERES project ends in December 2016.

An extension has been proposed for the current programme. Especially two suggested topics seem top be suitable and useful from the Apros validation point of view and VTT will participate in the related calculation work. The first one is a integral BWR containment system tests including the pool stratification phenomena. Due to complexity of the test some scoping calculations from the participants will be required for the test definition. The second one is a spray nozzle ring component test where the nozzles are embedded in a ring. The ring configuration represents more realistically the spray configuration of a real plant than in many previously made single nozzle test, and possibly reveals droplet interactions with the wall and with each other.

2.2.4 FONESYS (T2.4)

Benchmark exercises related to cold-injection waterhammer experiments and dryout experiments are planned within the FONESYS network. At least the previously postponed waterhammer benchmark is expected to be carried out during 2017. The organized benchmark exercises are analysed with Apros and reported to the organizers. In general the FONESYS activities are publicly reported through international conferences or scientific journal articles.

2.2.5 U. S. NRC CAMP In-Kind contribution (T2.5)

The current 2012-2017 U. S. NRC CAMP agreement requires two in-kind assessment cases to be calculated, reported and submitted to the U. S. NRC to be published in the NUREG series. As both of the current agreement period’s in-kind cases are currently unfulfilled it was decided in SAFIR2018 RG4 meeting 2/2016 that LUT and VTT will each cover one case before the agreement period ends in September 2017. As VTT currently has no suitable TRACE calculations that could be converted into in-kind contributions, the contribution is calculated and report is prepared in this task.

The in-kind contribution assessment is made of a topic that concerns a characteristic feature of a operating or planned power plant or benefits the validation of Apros by means of code-to-code comparison calculation. U. S. NRC CAMP agreement is covered in more detail in Chapter 2.3.3.
2.3 Management and international cooperation (WP3)

Work package 3 contains all non-research work in the project, i.e. administrative work and all work related to in various international nuclear-safety related activities. The main content of the work package is management of the project, participation in and possible arrangement of international project meetings and conferences, as well as possible document preparation carried out related to the international activities. The work package may also include attending training courses related to nuclear safety research for education of younger experts.

Partners and person months allocated to WP3 are listed in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>3.0</td>
</tr>
</tbody>
</table>

2.3.1 Management (T3.1)

This task consists of management of the project, including reporting to the reference groups, producing the annual reports, and attending the reference group meetings.

2.3.2 PKL-4 (T3.2)

OECD/NEA PKL-4 programme investigates safety issues relevant for current PWR plants as well as for new PWR design concepts by means of systematic parameter studies on thermal-hydraulic phenomena and transient tests under postulated accident scenarios. Participation in the program has been authorized by MEAE 15.6.2016 and Mr. Ismo Karppinen from VTT has been appointed as a member of the programme review group (PRG). The programme ends in May 2020. The programme is expected to hold two PRG meetings a year. This tasks contains the costs of participating in the meetings and distributing the related information to SAFIR2018 organisations.

2.3.3 CAMP (T3.3)

The current U.S. NRC’s Code Application and Maintenance Program (CAMP) agreement period covers years 2012-2017. The program is formed to exchange information on thermal-hydraulic safety related issues between U.S. NRC and its international partners. TRACE, PARCS, RELAP5 codes and graphical interface SNAP are made available through this program. Mr. Seppo Hillberg from VTT acts as Finland’s representative in the program. Expenses from participation in meetings as well as code distribution and communication between the parties are covered by this task. SAFIR2018 framework plan instructs collaboration through U.S. NRC’s CAMP program to be attached to a suitable TH-project. COVA project is perfectly suited to house this collaboration. The current CAMP program has a yearly participation fee of 25 k$ which is placed under WP4.

Negotiations for agreement period 2017-2022 are expected to begin around turn of the year 2016-2017. According to U.S. NRC international relations team, the cost of the new agreement period is likely stay the same as in the current period – 25 k$/a and 2 in-kind contributions during the 5 year period. For the purposes of this SAFIR2018 application, it is assumed that Finland continues participating in the program and the aforementioned expected participation costs are correct.

2.3.4 HYMERES (T3.4)

The current The OECD/NEA Hydrogen Mitigation Experiments for Reactor Safety (HYMERES) project ends in December 2016. The project has been holding two project meetings each year and Mr. Ismo Karppinen from VTT has been a member of the program review group (PRG), as appointed by the SAFIR2014 steering group. Yearly participation fee for the project has been 15 k€.
Extension for the project is being proposed. For the purposes of this SAFIR2018 application it is assumed that
the project is participated, PRG representation comes from VTT and that the number of yearly meetings and
participation fee stay the same. This task will cover participation of the meetings and distributing the gained
knowledge to SAFIR2018 organizations. Project participation fee is placed under WP4.

2.3.5 **WGAMA (T3.5)**

OECD/NEA CSNI Working Group on Analysis and Management of Accidents (WGAMA) aims to advance the
current understanding of the physical processes related to reactor safety. It holds one meeting each year. Mr.
Ismo Karppinen from VTT has been appointed as one of the two country representatives of Finland in the task
group.

2.3.6 **FONESYS (T3.6)**

The international network of system-thermal hydraulic code developers FONESYS (Forum & Network of Sys-
tem Thermal-Hydraulics Codes in Nuclear Reactor Thermal-Hydraulics) has been created to promote the use of
system thermal-hydraulic codes and application of Best estimate plus uncertainty approaches, to establish ac-
cceptable and recognized procedures and thresholds for verification and validation, and to create a common
ground for discussing envisaged improvements in system codes. Among its activities, FONESYS arranges
benchmarks for code comparison and verification and provides a channel of discussion for top level experts of the
system thermal hydraulic field. Participation in the FONESYS network is seen as an extremely beneficial tool for
developing and maintaining know-how related to system codes, and provides valuable support for the main object
of the COVA project. Currently Areva (France), CEA (France), KAERI (Korea), UniPi (Italy), and SNPTC (China)
participate in FONESYS in addition to VTT.

The internal documents and working material of FONESYS are in general not public. This is inevitable so that
proprietary data and material related to system-code development can be discussed freely within the network.
However, FONESYS aims at publishing all its main findings in scientific publications; the first FONESYS docu-
ment was published in Nuclear Engineering and Design in 2015, and the network’s activities have also been pre-
sented at NURETH-16 and NENE-24 conferences. A second FONESYS document related to the hyperbolicity
issue is to be published in 2017.

The proposed scientific activities within the FONESYS network to be carried out in the future include follow-up
of the critical flow benchmark with a large public data set from Université Catholique de Louvain (UCL), compari-
sion of CCFL (counter current flow limitation) treatment in system codes, and performing a comparative study on
scalability of closure laws. The next FONESYS meeting will be held during the summer of 2017 in Pisa.

This task includes the costs of participating in the meeting(s) and in document preparation done within the
FONESYS network. Actual research work done related to the FONESYS activities is covered by task T2.4 in
WP2. The annual participation fee of 8 k€ to the FONESYS is included in task T3.6 since this participation is not
entitled explicitly by MEAE and SAFIR2014 Steering Group to the predecessor of COVA, unlike the U. S. NRC
CAMP and the OECD/NEA projects.

2.3.7 **Training courses (T3.7)**

As instructed by SG2, this task has been left out of the 2017 project plan.
2.4 International participation fees (WP4)

Work package 4 contains the participation fees of U.S. NRC CAMP and the OECD/NEA projects. Project descriptions are presented in Chapter 2.3.

The total number of person months in WP4 is 0.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>0</td>
</tr>
</tbody>
</table>

2.4.1 Participation fees (T4.1)

In the year 2017 the costs funded through task T4.1 are the participation fees of U.S. NRC CAMP and the expected participation cost of OECD/NEA HYMERES.

<table>
<thead>
<tr>
<th>Year</th>
<th>Participation fee</th>
</tr>
</thead>
<tbody>
<tr>
<td>OECD/NEA HYMERES</td>
<td></td>
</tr>
<tr>
<td>2017</td>
<td>15 k€</td>
</tr>
<tr>
<td>2018</td>
<td>15 k€</td>
</tr>
<tr>
<td>2019</td>
<td>15 k€</td>
</tr>
<tr>
<td>2020</td>
<td>15 k€</td>
</tr>
<tr>
<td>U.S. NRC CAMP</td>
<td></td>
</tr>
<tr>
<td>annually</td>
<td>25 k$</td>
</tr>
</tbody>
</table>
## 3. Deliverables and milestones 2017

The planned deliverables and milestones for 2017 are listed in the table below.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.3.1</td>
<td>Report or a master's thesis on validation of Apros’ TH model against selected basic separate effect tests. If a diploma worker is recruited, a presentation of the work conducted so far will be given in RG4 meeting 3/2017, as suggested by RG4 in meeting 1/2017 (deliverable type &quot;other&quot;). If no suitable diploma worker can be found then one TH analysis case will be reported before 30.10.2017.</td>
<td>8.0</td>
<td>30.10.2017</td>
</tr>
<tr>
<td>D1.4.1</td>
<td>Report on validation of Apros’ containment model against a selected test. The case will be calculated and reported before 30.10.2017.</td>
<td>2.0</td>
<td>30.10.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Report on analysis of a PKL-4 experiment with Apros</td>
<td>1.5</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D2.3.1</td>
<td>Report on analysis of a HYMERES experiment with Apros</td>
<td>1.5</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D2.4.1</td>
<td>Report on Apros benchmark analyses conducted within the FONESYS network</td>
<td>1.0</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D2.5.1</td>
<td>U.S. NRC In-kind contribution report (NUREG-series publication). The report will be sent for review before 30.10.2017, actual approval date will depend on U. S. NRC.</td>
<td>1.0</td>
<td>30.10.2017</td>
</tr>
<tr>
<td>NA</td>
<td>Planning the Apros TH validation roadmap, updating the list of list of Apros validation cases and acquiring relevant experiment data.</td>
<td>0.25</td>
<td>NA</td>
</tr>
<tr>
<td>NA</td>
<td>Planning the Apros containment validation roadmap, updating the list of list of Apros validation cases and acquiring relevant experiment data.</td>
<td>0.25</td>
<td>NA</td>
</tr>
<tr>
<td>NA</td>
<td>Project management and cooperation in the following international programmes:</td>
<td>3.0</td>
<td>NA</td>
</tr>
<tr>
<td></td>
<td>- OECD/NEA PKL-4</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- OECD/NEA WGAMA</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- OECD/NEA HYMERES</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- U. S. NRC CAMP</td>
<td></td>
<td></td>
</tr>
<tr>
<td>• FONESYS network</td>
<td>Total pm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>-------------------</td>
<td>---------</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>18.5</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
4. Project organisation

The project is carried out as a whole at VTT Technical Research Centre of Finland, within the research team of Nuclear power plant behaviour. The project manager is Research Scientist, M.Sc. (Tech.) Mr. Seppo Hillberg. Senior Scientist, M.Sc. (Tech.) Mr. Ismo Karppinen will act as the deputy project manager and tutor to Mr. Hillberg in project management. The main researchers, the tasks they will be contributing to, and the estimated person months in 2017, are listed in the table below.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Joona Leskinen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.1, T3.7</td>
<td>1.5</td>
</tr>
<tr>
<td>Torsti Alku</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.4</td>
<td>1.0</td>
</tr>
<tr>
<td>Seppo Hillberg</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.5, T3.1, T3.3</td>
<td>2.0</td>
</tr>
<tr>
<td>Jarno Kolehmainen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T1.4, T2.3, T3.4</td>
<td>2.0</td>
</tr>
<tr>
<td>Eric Dorval</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.5</td>
<td>1.0</td>
</tr>
<tr>
<td>Ismo Karppinen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1, T1.3, T3.4, T3.5</td>
<td>1.5</td>
</tr>
<tr>
<td>Ari Silde</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.2, T1.4</td>
<td>1.25</td>
</tr>
<tr>
<td>Joona Kurki</td>
<td>Research Team Leader</td>
<td>VTT</td>
<td>T3.1, T3.6</td>
<td>0.25</td>
</tr>
<tr>
<td>Trainee N.N.</td>
<td>Research Trainee</td>
<td>VTT</td>
<td>T1.3</td>
<td>8.0</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>18.5</strong></td>
</tr>
</tbody>
</table>
5. Risk management

Greatest risks in this project are related to availability of the experimental data needed in WP1, and referred to in the OECD/NEA validation matrices. While lot of the data should be easily available through the NEA databank, a large part of the data is in the sole possession of the organizations that have performed the experiments, and may be difficult, costly or impossible to obtain; after all, some of the organizations that in the past have carried out such experiments have since ceased to exist. In these cases effort is put into trying to obtain alternative comparable data sets from other sources. In this effort, international cooperation is likely to prove invaluable, and help in obtaining further data may be available for example through OECD/NEA contacts or the FONESYS network.

Another risk to COVA project relates to cooperation projects and their progress. COVA intends to use input uncertainty quantification methods suggested by SAFIR2018 USVA project. As this project has been suffering from budget cuts, the results expected may not be ready for use during the project period of SAFIR2018.

As large portion of the work done in COVA is done by young researchers it is imperative to be successful in knowledge transfer. Effective guidance must be a priority and the methods applied in constant evaluation.
References


[11] Agreement on the OECD/NEA PKL Phase 4 project. To address thermal hydraulic safety issues for current PWR and new PWR design concepts through experiments in the integral test facility PKL (final draft 12.8.2016)
### Comprehensive and systematic validation of independent safety analysis tools

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Volume</td>
<td>Personnel</td>
</tr>
<tr>
<td>WP1 - Validation matrices</td>
<td></td>
<td></td>
</tr>
<tr>
<td>T1.1 Critical assessment of TH model's validation</td>
<td>0.25</td>
<td>3.8</td>
</tr>
<tr>
<td>T1.2 Critical assessment of containment model's validation</td>
<td>0.25</td>
<td>3.8</td>
</tr>
<tr>
<td>T1.3 Validation analyses of TH model</td>
<td>8.0</td>
<td>52.5</td>
</tr>
<tr>
<td>T1.4 Validation analyses of containment model</td>
<td>2.0</td>
<td>25.0</td>
</tr>
<tr>
<td>WP2 - Analyses of new experiments</td>
<td>5.0</td>
<td>61.0</td>
</tr>
<tr>
<td>T2.1 PKL-4</td>
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<td>18.0</td>
</tr>
<tr>
<td>T2.2 Support of LUT PWR depressurization</td>
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<td>0.0</td>
</tr>
<tr>
<td>T2.3 HYMERES</td>
<td>1.5</td>
<td>18.0</td>
</tr>
<tr>
<td>T2.4 FONESYS</td>
<td>1.0</td>
<td>13.0</td>
</tr>
<tr>
<td>T2.5 CAMP In-kind contribution</td>
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<td>12.0</td>
</tr>
<tr>
<td>WP3 - Management and international cooperation</td>
<td>3.00</td>
<td>41.0</td>
</tr>
<tr>
<td>T3.1 Management</td>
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</tr>
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<td>T3.2 PKL-4</td>
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<tr>
<td>T3.4 HYMERES</td>
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<td>7.0</td>
</tr>
<tr>
<td>T3.5 WGAMA</td>
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<td>4.0</td>
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<tr>
<td>T3.6 FONESYS</td>
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<td>3.0</td>
</tr>
<tr>
<td>T3.7 Training courses</td>
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<td>0.0</td>
</tr>
<tr>
<td>WP4 - International participation fees</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>T4.1 Participation fees</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>TOTAL</td>
<td>18.50</td>
<td>187.0</td>
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**Comments:**

WP1 tasks 1&2 continue in 2017 with low volume. Refining and filling the gaps of the roadmap is needed when the work is on-going. Data for suitable experiments is gathered.

Volume of WP3 is necessarily relatively large with respect to the total volume of the project due to the large number of projects involved there.

WP3 international cooperation tasks all contain the costs of participating in one or two meetings per year, depending on the project.

WP4 contains direct participation fees of international projects with full VYR funding (CAMP 25 k€ and the expected fee of HYMERES 15 k€).

3.2.2017: T3.7 has been left out.

3.2.2017: Due to the corresponding task being left out of SAFIR2018 INSTAB, T2.2 will not actualize.

3.2.2017: Budget cut being 20 k€, the resulting 4.5 k€ has been moved to T1.3 to be used on a master's thesis.

3.2.2017: T1.3 person months has been increased to reflect diploma worker's (trainee's) salary.
SAFIR2018 Project plan

INSTAB

Couplings and instabilities in reactor systems

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1. Research theme and motivation

There are several scenarios of safety importance where containment pressure suppression function and pressure suppression pool (PSP) operation are affected by (i) stratification and mixing phenomena, (ii) interactions with emergency core cooling system (ECC), spray, residual heat removal (RHR) and filtered containment venting (FCV) systems, (iii) overall water balance in the containment compartments, and (iv) interplay between pool behaviour, diagnostics and procedures. Specifically those scenarios include (i) different LOCAs including scenarios with steam line break inside the radiation shield, broken blowdown pipes, and leaking safety relief valves, (ii) station blackouts, and (iii) severe accidents. There is a need for validated tools for simulation of realistic accident scenarios with interplay between phenomena, safety systems, operational procedures, and overall containment performance.

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. Addressing stratification and mixing issues in a large pool is thus important and additional data on pool behaviour are needed for validation of computer models and realistic evaluation of safety margins [1].

1.1 Background and state-of-the-art

In the international evaluation of the Finnish Nuclear Safety Research Programme SAFIR2014 the panel recommended that CFD methods for solving two phase flow problems should be validated against experiments. The importance of the validation of the codes is thus emphasized in the SAFIR2018 Framework Plan. Also OECD/NEA working groups have expressed that CFD tools have to be comprehensively validated before they can be used in licensing calculations of NPPs. Therefore, CFD grade validation data, that can be obtained, for instance, with Particle Image Velocimetry (PIV), wire-mesh sensors (WMS) or high speed cameras, is needed.

The three Roadmaps of the Nordic Thermal Hydraulic Network (NORTHNET) focus on the development of numerical thermal hydraulic methods and performing experiments. The recently revised Roadmap 3 deals with experiments and modelling of stratification in pressure suppression pools of BWRs [2]. Also, studies on safety relief valve (SRV) spargers, residual heat removal (RHR) system nozzles and strainers as well as on drywell and wetwell sprays are considered important and proposed in the roadmap.

BWR containment is a complex system that includes such typical elements as a pressure suppression pool, spray and containment venting systems for containment pressure control, blowdown pipes for rapid steam condensation in case of LOCA, spargers for the vessel safety relief valves (SRV), strainers for water supply to emergency core cooling and spray systems, nozzles and strainers of the RHR system, vacuum breakers, etc. Dynamic loads due to the direct contact condensation (DCC) and thermal stratification and mixing induced by steam injection through a blowdown pipe have been studied in the previous SAFIR projects [3, 4, 5, 6, 7]. In the INSTAB project the phenomena that can affect pressure suppression function due to the operation of the other equipment and systems in the BWR containment will be investigated.

Particularly the aim is to experimentally study the interplay between pool behaviour and spray systems, the effect of RHR system and SRV sparger operation on pool mixing and inverse temperature stratification due to injection of cold spray water and falling liquid films. Although several full scale experiments have been done on wetwell pool mixing due to pressure relief system blowing and activation of systems for forced mixing, limited high-grade data is available, for example, on the details of pool mixing due to activation of the wetwell spray systems and on the effects of falling liquid films. Additional data of pool interactions with spray systems are needed for the realistic evaluation of safety margins in designs and for validation of computer models.

The existing database of suppression pool tests of LUT contains experiments that require in-depth analysis to reach eventual conclusions of their results. The most interesting ones of these tests are the tests with a blowdown pipe collar [8] and with parallel and transparent (poorly thermally conducting) blowdown pipes [9]. Recent and ongoing work on CFD code and model development and increased computational capacity make the simulations of these cases appealing i.e. expected simulation results would at least qualitatively mimic the experimental reality.
Thus the CFD simulations and other emerged analysis methods, such as pattern recognition algorithms, could be used both to understand the experimental results and to validate the computational models with these special test cases. Note: In 2017 these CFD calculations will be done under the NURESA project managed by VTT.

The Effective Heat Source (EHS) and Effective Momentum Source (EMS) models for steam injection through blowdown pipes have been successfully developed, validated and implemented to the GOTHIC code by KTH with the help of the experiment results of the previous SAFIR and NKS projects [11]. The experiments to be carried out in the INSTAB project and complementary CFD simulations at VTT in the NURESA project will help further extend the concepts of the EHS and EMS models to spargers, strainers, RHR system nozzles, and operation of blowdown pipes with noncondensible gases. Work to extend the models to cover the phenomena related to SRV spargers is underway on the basis of sparger tests carried out in the INSTAB project in 2015-2016 [12, 13]. The possibilities to implement the EMS and EHS models into APROS containment code will also be reviewed.

The BWR containment behaviour is affected by the interaction of noncondensible gas (containment atmosphere) and the liquid / steam flows (initial gas bubbles in the suppression pool, spray droplets dissolving noncondensible gas, noncondensible venting from wetwell gas space back to drywell, and noncondensible gas effects on heat exchangers (if present)). Thus PPOOLEX provides an excellent platform to study the dynamics of noncondensible gas dissolution and release. Studies will be started by a survey of the existing experiment data and models. Depending on the availability and quality of data, experiments in LUT may be proposed later. The target of this review is to explore the potential for significant improvement in the physical modelling of noncondensible gas behaviour in system codes, containment codes, and CFD codes.

Ultimately, the proposed INSTAB project would advance the comprehensive validation work of CFD codes beyond the state-of-the-art by providing high-grade measurement data of instability phenomena from selected scenarios of safety importance in large fluid volumes such as suppression pools. With the help of the experiment results obtained in the project, CFD tools will be developed in the co-operating research organizations for the modelling of spargers, RHR system nozzles, spray operation and formation of liquid films on the vessel wall and blowdown pipe. Methods for the estimation of stratification of the water pool during the operation of sprays are developed. Capability of the APROS lumped-parameter containment code in modelling containment spray cooling and mixing of water pool during the spray injection in BWR containment conditions will be evaluated.

1.2 Objectives and expected results

The SAFIR2018 Framework Plan states that the objective of the research programme 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 these matters.

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 will be gathered on

- effect of SRV spargers, RHR nozzles, strainers and blowdown pipes on mixing and stratification of the pool;
- 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;
- formation of liquid films on the vessel wall and blowdown pipe due to spray operation and their effect on heat transfer and local condensation and heat flux to the pool;
- earlier suppression pool test results concerning blowdown pipe with a collar, parallel blowdown pipes and transparent/poorly conducting blowdown pipes

To achieve the objectives a combined experimental/analytical/computational program will be carried out. LUT will create an experiment database on pool operation related phenomena in the PPOOLEX test facility and in a small scale separate effects test facility with the help of sophisticated, high frequency measurement instrumentation and high-speed video cameras. LUT, VTT and KTH will use the gathered experiment database for the development, improvement and validation of numerical simulation models. Also analytical support will be provided for the experimental part by pre- and post-calculations of the experiments.

Expected results from the whole duration of the project:

1) The EMS/EHS models will be developed and validated for blowdown pipes, SRV spargers, RHR nozzles, and spray system operation in the wetwell. The development and validation work of the EMS/EHS models will be done by KTH on the basis of the experiment results obtained at LUT in the INSTAB project. The models will be available to be implemented to CFD and system codes for the organisations involved in the SAFIR2018/RG4 work at the end of the research program.
2) CFD calculation models will be improved for the simulation of steam injection through a SRV sparger into the suppression pool and for spray water injection into the wetwell gas space. These CFD models will be developed by VTT on the basis of the experiment results obtained at LUT in the INSTAB project.

3) OpenFOAM models for the simulation of direct contact condensation phenomenon will be developed and validated against INSTAB experiments results at LUT and VTT.

Expected results from 2017:

1) Wetwell spray experiments will reveal if mixing of a thermally stratified pool with the help of spray injection from above is successful.

2) The sparger test series in PPOOLEX will be concluded with a final experiment and closures for the EMS model development work for spargers will be provided.

3) Effective momentum induced by steam injection through a SRV sparger will be directly measured in a small scale separated effect test facility.

4) Survey of noncondensible gas dissolution/release dynamics modelling will be provided.

The main benefit of the project will come through improved and validated calculation models of CFD and system codes used for nuclear safety analysis. The project outcome will allow the end users to analyse the risks related to different scenarios of safety importance in the containment of a Nordic BWR. The research results can be used by the power companies, nuclear safety authorities and research organizations.

1.3 Exploitation of the results

The experiments carried out by LUT will provide high quality measurement data which can be used to improve the analytical tools used in the safety analyses of nuclear power plant systems. With the help of the data the models in system and CFD codes and in other analytical methods can be validated. Furthermore, the EMS and EHS models, developed and validated at KTH on the basis of the LUT experiments, will be available to be implemented in the APROS containment code for the calculation of phenomena related to pool stratification and mixing.

The ultimate end users of the project results will be the Finnish and Swedish safety authorities, power utilities and research organizations using CFD and system codes improved and validated with the help of the experiment data provided by the INSTAB project.

Each individual developed and improved calculation model can be implemented in the simulation tools as soon as the validation effort against the experiment results provided by the project has been finished. Many of the experiment results from 2015-2016 have already been applied to the development work of the EMS/EHS models for SRV spargers and RHR nozzles as well as to the development work of CFD models for spargers and sprays.

1.4 Appropriateness of the project to SAFIR2018 programme

The SAFIR2018 Framework Plan states that the thermal hydraulic computational software widely used in safety analyses must be validated by using experimental data that reflect the key characteristics and operating parameters of Finnish plants. National infrastructure and research teams are needed to produce the data. The teams must have the ability to build and operate experimental equipment that accurately represents the phenomena. Existing equipment can be modified and, where research needs stipulate, the know-how for building new equipment must also exist.

The INSTAB project enhances experimental, analytical and computational methods employed by the research community in solving safety problems of nuclear power plants. The existing experiment facilities at LUT will be modified and new testing equipment will be designed and built to be used in studies related to Nordic BWR containments. An experiment database will be gathered and utilized in the development and improvement work of simulation tools both in the INSTAB and other SAFIR2018 projects.

According to the Framework Plan the research programme will create competence, joint activities and networking. The combined research effort of LUT, VTT and KTH in the INSTAB project will support these goals by creating a research group, where the experiment results obtained at LUT will be utilized by others in the development work of calculation methods and tools. The proposed INSTAB project is part of the national nuclear safety research effort that develops and creates expertise, test facilities and computational methods to be used to solve nuclear safety related problems.
1.5 Education of experts

The SAFIR2018 Framework Plan states that the safe use of nuclear power in Finland can be ensured only with the help of high-quality national expertise. The INSTAB project aims to strengthen the expertise related to designing, constructing and operating test facilities used for modelling the behaviour of safety related systems of NPPs. Expertise on sophisticated measurement and visual recording techniques, such as PIV, WMS and high speed cameras, in demanding thermal hydraulic conditions will increase during the project. Two members of young generation, M.Sc. Lauri Pyy and M.Sc. Joonas Telkkä will specialize on the use of the PIV measurement system both in gaseous atmosphere and in volumes filled with fluid. M.Sc. Elina Hujala will utilize the high speed camera and field data of the experiments in her dissertation dealing with interfacial area transport and pattern recognition algorithms. Dmitry Skripnikov’s master’s thesis deals with the measurement of spray droplets using the shadowgraphy system. A master’s thesis for double degree studies on steam sparger modelling for boiling water reactor suppression pool will be written under the INSTAB project.
2. Work plan

Research in the INSTAB project focuses on such scenarios of safety importance where the containment pressure suppression function and pressure suppression pool operation are affected by stratification and mixing phenomena or interactions between the ECC, spray, SRV sparger and RHR systems. The overall strategy is to co-operate closely with the simulation partners VTT and KTH in defining test conditions and procedures in order to provide them with measurement data for the development and improvement of calculation tools. The project is planned to last from 2015 to 2018 as is also the case with the related COPSAR (Containment Pressure Suppression Systems Analysis for Boiling Water Reactors) project which is funded through NKS. Pre- and post-test calculations of thermal hydraulic experiments in the INSTAB project will be done with CFD and system codes by VTT in the NURESA and COVA SAFIR2018 projects dealing with development and improvement of simulation tools. The research plan of the INSTAB project has been written on the basis of recommendations received from the SAFIR2018 Reference Group 4 and NORTHNET RM3 reference group. The planned costs and funding for the whole four year period are indicated in Annex 3.

The schedule of the INSTAB work packages and tasks for 2017-2018 is presented in the table below.

<table>
<thead>
<tr>
<th>INSTAB, Work packages and Tasks</th>
<th>2015</th>
<th>2016</th>
<th>2017</th>
<th>2018</th>
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<tbody>
<tr>
<td>WP1 - SRV sparger tests</td>
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<tr>
<td>T1.1 A test with the sparger in the center position</td>
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<td>T1.2 A small scale separate effect test facility for sparger studies</td>
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<tr>
<td>T1.3 Tests with the small scale separate effect test facility</td>
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<td>WP2 - RHR system tests</td>
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<tr>
<td>T2.1 Mixing tests with a RHR nozzle</td>
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<tr>
<td>WP3 - Effect of noncondensible gases on DCC</td>
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<td>T3.1 Blowdown pipe tests with noncondensible gases</td>
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<td>WP4 - Spray studies</td>
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<td>T4.1 Wet well spray tests</td>
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<td>T4.2 Experiments on the behaviour of liquid film</td>
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<td>WP5 - CFD calculations of earlier tests</td>
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<tr>
<td>T5.1 Simulation of the collar blowdown pipe case</td>
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<td>T5.2 Simulation of the sparger pipe case</td>
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<tr>
<td>WP6 - Dissolution and release of noncondensible gas</td>
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<td>T6.1 Initial survey of models and data</td>
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<td>T6.2 Planning of test arrangements</td>
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<td>WP7 - Project management</td>
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<tr>
<td>T7.1 Project management, Nordic co-operation and publications</td>
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A short summary of the content of each work package for 2017 is presented below. A more detailed presentation of the research to be carried out in 2017 can be found in following chapters.

In Work Package 1, a SRV sparger test, where the SRV sparger will be first moved to the centre of the pool and the submergence reduced from 1.8 to 1.5 m, will be carried out to complete the experiment series with the SRV sparger in the PPOOLEX facility. A small scale separate effect facility, where it is possible to measure directly the effective momentum induced by a steam injection through a single hole for different condensation regimes, will be designed and constructed. Tests with the facility will provide closures for the EMS model development for spargers by KTH.

In Work Package 2, experiments on mixing efficiency due to water injection through a RHR nozzle have been carried out in PPOOLEX in 2016 to provide validation data for KTH to be used in the extension of the EMS and EHS models. Conditions leading to complete mixing have been determined in cases where the RHR nozzle is either vertically or horizontally oriented. The work package is finished.

In Work Package 3, effect of noncondensible gases on direct contact condensation in case of steam discharge through a blowdown pipe will be investigated. The tests will further extend the concept of the EHS and EMS models. These studies are scheduled for 2018.
In Work Package 4, spray injection experiments in PPOOLEX will be carried out. Interplay between suppression pool behaviour and the spray system will be addressed and verification data for improving simulation models in system and CFD codes at VTT and KTH will be provided.

In Work Package 5, a CFD calculation of the earlier experiment with a collar shaped blowdown pipe outlet will be performed. The goal is to exploit the extensive database gathered in the previous PPOOLEX studies of steam discharge into a pool of sub-cooled water in the assessment of the capability of CFD codes to simulate direct contact condensation situations. The simulation of the test includes also an effort to analyse the experimental results further in order to draw more conclusions on them. *Note: Due to reductions to the applied funding this WP will be included in the NURESA project instead of the INSTAB project in 2017.*

In Work Package 6, LUT will conduct a survey of the existing experiment data and calculation models for non-condensible dissolution/release dynamics. Depending on the availability and quality of data, experiments in LUT may be proposed later.

### 2.1 SRV sparger tests (WP1)

Sparger tests in PPOOLEX have provided data for the extended development of the EMS model. In the tests performed in 2015-2016 steam injection was either through the sparger head or the LRR. Mixing of a thermally stratified water pool was successful with a quite small steam flow rate when the flow direction was downwards i.e. through the LRR. Complete mixing was achieved also via radial injection through the sparger head if the steam flow rate was higher. The movement of the thermocline during steam injection via the sparger pipe was studied in detail, too.

The sparger test series in PPOOLEX will be completed in 2017 with the sparger first moved to an alternative position, centre of the pool, and the submergence reduced from 1.8 to 1.5 m. With the help of the experiment results further development of the EMS model for SRV spargers will be pursued to simulate dynamics of the pool mixing and stratification.

A small scale separate effect facility equipped with a sparger having only one injection hole will be designed and constructed at LUT. Effective momentum induced by steam injection through a single hole for different condensation regimes will be measured directly. Experiments with this facility will give the additional data on sparger behaviour which is needed to provide closures for the EMS model development work for spargers.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
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<tr>
<td>LUT</td>
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</table>

#### 2.1.1 A test with the sparger in the centre position (T1.1)

A SRV sparger test, where the sparger is first moved to the centre of the pool and the submergence is reduced from 1.8 to 1.5 m, will be carried out in the PPOOLEX facility. This would allow development of a thicker stratified layer at the bottom and would also contribute to the EHS/EMS models based on the Richardson scaling. Test specifications will be decided together with KTH on the basis of earlier tests and a pre-test simulations. The aim is to define momentum fields in the pool caused by injection of steam through the small holes directed radially at the sparger head.

Instrumentation of PPOOLEX needs to be slightly modified in order to capture the effect of the reduced submergence on stratification/mixing. Particularly the location of the thermocouple measurements in the pool volume need to be changed.

#### 2.1.2 A small scale separate effect test facility for sparger studies (T1.2)

A small scale separate effect test facility, where it is possible to measure directly the effective momentum induced by steam injection through a single hole for different condensation regimes, will be designed and constructed at LUT. The facility will consist of a small pool with transparent walls and of a sparger pipe having a single injection hole with a diameter similar to the holes in the PPOOLEX sparger. Figure 1 illustrates the main operational principle of the proposed facility. Effective momentum will be evaluated with the help of a direct torque measurement. High speed cameras will allow recordings of the condensation regimes and collapsing bubbles. With high frequency pressure measurements the detachment and collapse frequency of the bubbles will be obtained.
2.1.3 Tests with the small scale separate effect test facility (T1.3)

A series of steam discharge experiments will be carried out in the small scale separate effect test facility. For example in oscillatory condensation regime, bubbles detach from the jet at 50-400 Hz. The effective momentum is expected to be dependent of the collapse mode. In “asymmetric” collapse momentum is believed to be transferred to the mean flow and turbulence but in “symmetric” collapse momentum is believed to be transferred to turbulence, not to the mean flow. PPOOLEX separate effect experiments will allow to measure and visualize directly this phenomena. The tests will thus help to map the effective momentum of many condensation regimes and will provide closures for the EMS model development for spargers by KTH.

2.2 RHR system tests (WP2)

Mixing efficiency due to water injection through a RHR nozzle have been studied in PPOOLEX in 2016. The PPOOLEX test facility has been equipped with a scaled down model of a RHR nozzle and an experiment series with the system has been carried out to provide validation data for KTH to be used in the extension of the EMS and EHS models. Conditions leading to complete mixing have been determined in cases where the RHR nozzle has been either vertically or horizontally oriented. A thermally stratified condition has been first created with small steam injection through the SRV sparger. Single phase water injection through the RHR nozzle has then been used to mix the pool. The effect of flow rate and water temperature on mixing efficiency has been studied. Data from the experiments has been delivered to the simulation partners and the work package is finished.

2.3 Effect of noncondensible gases on DCC (WP3)

Modifications of the CFD and EHS/EMS models for the blowdown pipes in case of different steam condensation regimes and presence of noncondensible gases need to be developed in order to have a complete set of models for simulation of the most important components in the Nordic pressure suppression system such as blowdown pipes, spargers, RHR nozzles, and sprays. A study on the effect noncondensible gas on thermal stratification and flow patterns in suppression pool has been done by Cai et al. but further experimental research on the issue is
needed [14]. For this purpose validation experiments on DCC in the presence of noncondensible gases will be conducted in the PPOOLEX facility equipped with a prototypical blowdown pipe in 2018.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
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2.3.1 Blowdown pipe tests with noncondensible gases (T3.1)


2.4 Spray studies (WP4)

Spray studies for improving simulation models in CFD and system codes will continue in 2017. LUT will carry out spray experiments in the PPOOLEX facility and deliver measurement data to simulation partners VTT and KTH to be used in model development work. Interplay between suppression pool behaviour and the spray system is of special interest.

In 2015, single spray nozzle tests with different capacity full cone nozzles were carried out in an open test environment in order to develop a measurement procedure for determining droplet size and velocity distributions of the spray jets [15]. The shadowgraphy application of the PIV measurement system was used. The measured single nozzle test data has been used for comparison of the preliminary CFD calculations of spray operation at VTT.

A spray injection system was constructed and installed to the wetwell compartment of the PPOOLEX facility during the last quarter of 2016 and preliminary wetwell spray tests were carried out at the end of 2016. In 2017 the spray experiments will focus on the interaction between the spray system and the wetwell pool, and in 2018 on the behaviour of liquid films formed as a result of spray operation.

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<thead>
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<th>Partners in WP4</th>
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2.4.1 Wetwell spray tests (T4.1)

Mixing of a thermally stratified pool with the help of spray injection from above is of interest. 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. With the help of pre-test simulations done at VTT and KTH a representative test case with a suitable initial thermal hydraulic state of the facility and a correct spray injection rate to be used can be determined. Smaller than applied SAFIR and NKS funding will reduce the number of planned tests. For example tests on spray nozzles of a different type or capacity may need to be postponed.

2.4.2 Experiments on the behaviour of liquid films (T4.2)


2.5 CFD calculations of earlier tests (WP5)

Recent and ongoing work on CFD code and model development and increased computational capacity make the simulations of earlier DCC experiments in the POOLEX and PPOOLEX facilities appealing. CFD simulations and other emerged analysis methods, such as utilization of pattern recognition algorithms, could also help to understand the experimental results more profoundly. The goal is to exploit the extensive database gathered in the previous PPOOLEX studies of steam discharge into a pool of sub-cooled water in the assessment of the capability of CFD codes to simulate direct contact condensation situations.

Simulations of the PPOOLEX DCC-05 case in 2014-2015 indicated that chugging mode in a vertical blowdown pipe cannot be reached computationally in an economic way with the same models which were promising in the open pool POOLEX simulations [16]. 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 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 in a simulation of a straight blowdown pipe DCC experiment [17, 18].

In 2016, a plexiglass blowdown pipe experiment done in PPOOLEX was simulated with 2D hexahedral mesh with NEPTUNE_CFD/OpenFOAM. The idea with the plexiglass experiments was the possibility to see inside the blowdown pipe to detect the movement of the steam/water interface. Furthermore, the heat conductivity of the pipe wall was different and thus it was possible to see its effect on the condensation phenomena as well. Numerical problems seem to be worse than in the smaller pipe DCC-05 case. Pattern recognition analysis of the plexiglass test was started during the last quarter of 2016.

In 2017, a test done with the collar shaped outlet of the blowdown pipe will be simulated with CFD. In 2018, a simulation model of the sparger pipe used in PPOOLEX could be developed and a representative experiment could be calculated. **Note: Due to reductions to applied funding this task will not be included in the INSTAB project at all in 2017. Instead, CFD calculations of earlier PPOOLEX tests will be included in the NURESA project manged by VTT.**

**CFD software policy in the project:**

The number of capable two-phase CFD software for rapid phase change simulations is limited. The main objective is to obtain simulation results that can be compared to corresponding test results. The results should be of high quality enough to be published in academic journals. Due to the license issues and the functionality of the available software, the code usage priority within the project is:

0. The simulations are started with OpenFOAM software. If the NEPTUNE_CFD license is not available, only OpenFOAM results will be reported.
1. If both the OpenFOAM and NEPTUNE_CFD codes are available, simulations with both of them are attempted and the results are reported. This is the optimal case for the sake of code and model development.
2. If the NEPTUNE_CFD license is available and the OpenFOAM simulation results are insufficient, only the NEPTUNE_CFD results will be reported in full extent.

Insufficient result means in this case, that the result is not converged or the successfully simulated transient is too short to be compared to the test data.

<table>
<thead>
<tr>
<th>Partners in WP5</th>
<th>Person months</th>
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</thead>
<tbody>
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### 2.5.1 Simulation of the collar blowdown pipe case (T5.1)

A simulation model of the blowdown pipe equipped with a collar shaped outlet in PPOOLEX will be constructed and a representative test case will be calculated with the CFD code. The simulation of the test includes also an effort to analyse the experimental results further in order to draw more conclusions on them. Particularly the pattern recognition analysis could help in this since there is a lot of high speed recordings from the collar pipe experiments. **Note: This task in transferred to the NURESA project.**

### 2.5.2 Simulation of the sparger pipe case (T5.2)


### 2.6 Dissolution and release of noncondensible gas (WP6)

Improving the modelling of noncondensible gas dissolution and release is of interest for all safety analyses, containment and reactor. Traditionally, multicomponent modelling has assumed that noncondensibles travel only with the steam phase. However, in reality, dissolution in water masses and release, also takes place, affecting system pressure (by reducing noncondensible partial pressure), flow rates (by bubbling at throttled sections and steam bubble formation in piping), and heat transfer (by blanketing heat transfer surfaces). The effects of dynamics of
noncondensible gas dissolution and release are at present poorly understood; however, building on the successful modelling of direct contact condensation mentioned in section 2.5, LUT is poised to develop a significantly improved physical model of noncondensible gas dynamics in system codes, containment codes, and CFD codes.

LUT is participating in an H2020 proposal NONCOND which aims to improve the understanding of noncondensible gas behaviour in the reactor system. Although flow configurations inside a reactor and inside a BWR containment differ from each other, basic physical phenomena are the same. Thus INSTAB WP6 links with NONCOND, and if this EU-project materialises, NONCOND and INSTAB/WP6 would build synergy between reactor system modelling and containment system modelling.

2.6.1 Initial survey of models and data (T6.1)

Studies will start in 2017 by a survey of the existing experiment data and models for noncondensible dissolution/release dynamics. Depending on the availability and quality of data, experiments in LUT may be proposed later. Due to reductions on the applied funding the survey may not be completed until in 2018.

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2.6.2 Planning of test arrangements (T6.2)


2.7 Project management (WP7)

<table>
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2.7.1 Project management, Nordic co-operation and publications (T7.1)

Participation and preparation of SAFIR2018 reference group meetings. Nordic-operation will be enhanced via participation to the work done in the NORTHNET framework and through a common NKS project with VTT and KTH. Scientific journal articles and conference papers on different research topics of the INSTAB project will be written together with simulation partners.

2.7.2 SAFIR2018 midterm seminar (T7.2)

Participation of the SAFIR2018 midterm seminar.
# 3. Deliverables and milestones 2017

<table>
<thead>
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<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
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<td>D1.1.1</td>
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<td>1.5</td>
<td>31.10.2017</td>
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<td>A test with steam injection through the sparger pipe in PPOOLEX. The sparger is moved to the</td>
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<td>centre position of the pool and submergence is reduced.</td>
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<td>Criterion for approval: Experiment successfully done.</td>
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<tr>
<td>D6.1.1</td>
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<tr>
<td>--------</td>
<td>-----------------------------------------------------------</td>
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<tr>
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</table>

2018

2.5 x.x.2018

2.0 31.12.2017
4. Project organisation

Mr. Markku Puustinen from LUT will act as the project manager. LUT / School of Energy Systems / Nuclear Engineering is responsible for the whole project.

As the project is planned to be carried out by LUT, there are no project partners. However, the research effort in the INSTAB project is closely connected to the work done by VTT in the SAFIR2018/NURES project and by KTH in the framework of NORTHERN RM3 and OECD/HYMERES project. Furthermore, the common NKS/COPSAR project of LUT, VTT and KTH deals with the same research topics as described in this work plan.

Since, this work is very much dealing with experiments, it is impossible beforehand to decide the exact working hours of a single person. Thus, only estimated person months are presented with the full list of persons who can be involved in the research.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
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<tr>
<td>Markku Puustinen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP3, WP4, WP6, WP7</td>
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<tr>
<td>Antti Räsänen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP3, WP4</td>
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<tr>
<td>Heikki Purhonen</td>
<td>Research director</td>
<td>LUT</td>
<td>WP7</td>
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<tr>
<td>Jani Laine</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP3, WP4</td>
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<td>Vesa Riikonen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP7</td>
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<tr>
<td>Joonas Telkkä</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP3, WP4</td>
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<td>Lauri Pyy</td>
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<td>Harri Partanen</td>
<td>Design engineer</td>
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<td>Eetu Kotro</td>
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<td>Elina Hujala</td>
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<td>Giteshkumar Patel</td>
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<td>Juhani Hyvärinen</td>
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</table>
5. Risk management

Project risks arise from potentially limited availability of human resources and from a slight possibility of a major equipment failure.

The experimental and theoretical work proposed here requires certain competences that only senior research staff possesses. In case of unavailability of the necessary knowledge, project scope may need to be reduced and/or emphasis shifted to areas which can be completed using available resources. The Project team could in such a case seek assistance of the stakeholders, because the Finnish SAFIR partners do possess much of the necessary knowledge.

Failures to complete experiments due to single hardware or other systemic issues will cause at worst a delay of a few months in completing the study. This applies to experiments both in WP1 and WP4. All experiments envisioned here are relatively large in size and complicated in nature and therefore cannot be repeated many times over. For this reason there is a slight risk of a Work Package failing completely due to a failure of one major facility or piece of equipment.

There are new users of electric power in LUT laboratories, it means that scheduling the experiments is now more challenging.

A catastrophic event such as a massive fire in the laboratory or at LUT campus is excluded from the risk assessment.
References


The Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2018)

Resource Plan for 2017

Author: Heikki Purhonen

### Expenses

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<th>Work packages and Tasks</th>
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<th>Personnel</th>
<th>Mat &amp; supp</th>
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Comments:
- WP2 has been completed.
- WP5 is left out due to reductions to applied funding
- No activity in tasks T3.1, T4.2 and T6.2 is planned for 2017.
SAFIR2018 Project plan

INTEGRA

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

Vesa Riikonen, Heikki Purhonen, Juhani Hyvärinen,
Lappeenranta University of Technology
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   1.2 Objectives and expected results
   1.3 Exploitation of the results
   1.4 Appropriateness of the project to SAFIR2018 programme
   1.5 Education of experts

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     2.1.1 Participating in the OECD/NEA PKL Phase 4 project (T1.1)
     2.1.2 Effects of nitrogen from an accumulator (T1.2)
     2.1.3 Flow reversal due to a pump trip (T1.3)
     2.1.4 Inadvertent opening of SV and MSRT (T1.4)
   2.2 Passive heat removal circuits (WP2)
     2.2.1 Fundamentals of passive systems (T2.1)
     2.2.2 Designing and construction of test environment of selected passive system (T2.2)
   2.3 Project management (WP3)
     2.3.1 Project management (T3.1)
     2.3.2 SAFIR2018 midterm seminar (T3.2)
     2.3.3 Publications (T3.3)
     2.3.4 OECD/NEA PKL Phase 4 project participation fees (T3.4)

3. Deliverables 2017

4. Project organisation

5. Risk management

References
1. Research theme and motivation

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. Performing experiments not only requires the hardware and programs controlling the devices and gathering data, but also the knowledge of the system behaviour. Computer analyses with system and CFD codes are needed in the planning of the experiments as well as in post analyses to help understanding the physics in the experiments.

Gaining expertise in performing thermal hydraulic tests and designing and constructing test facilities connected with the ability to make analyses with computer codes answers directly to the mission of developing and creating expertise, test facilities and computational methods for solving safety problems of nuclear power plants.

1.1 Background and state-of-the-art

This project is formed of the Work Packages dealing with integral tests with PWR PACTEL and passive heat removal circuit.

A part of the international efforts in enhancing the reactor safety is the OECD projects. Finland participates several of such projects as the OECD/NEA PKL Phase 4 project and also provided test data to the OECD/NEA PKL Phase 3 project. The most of the OECD countries using nuclear power are participating the PKL projects. The OECD/NEA PKL Phase 4 experiments will investigate safety issues relevant for current PWR plants as well as for new PWR design concepts by means of systematic parameter studies on thermal-hydraulic phenomena and transient tests under postulated accident scenarios. The proposed test topics have been arranged according to the two focus areas implemented into the experiments: parametric studies on thermal-hydraulic procedures for model development and validation of T/H system codes and experimental verification of cool-down procedures and operation modes for different incidents and accidents.

The first category addresses test subjects related to current safety issues that either suffer from the lack of a dedicated data base for analysis and validation of computer codes or from uncertainties in the safety evaluation stemming from open issues or questions. The extension to already existing data bases related to these subjects is the foremost goal of this first category experiments. The second category of tests mostly contains transient tests either on test subjects already investigated in the former OECD/PKL-projects as answers to questions that could not yet finally be completed or on subjects which represent current topics from the international debate on PWR safety. Finland and LUT provides two integral tests (impact of nitrogen on cool-down/heat removal) performed with the PWR PACTEL facility in four years duration of the project.

Tripping of a reactor coolant pumps causes asymmetric flow conditions in the primary loops as well as in the reactor core. In an ultimate case the decreased coolant flow in part of the core leads to decreased heat transfer from the fuel rods to the coolant. A pump trip initiates a partial trip (PT) signal and closure of the turbine valve followed by the load reduction. A flow reversal occurs in the affected loop due to reversal of the pressure distribution in this loop caused by the other running pumps. This results backflow in the loop with the idle reactor coolant pump and slight overflow in the other loops. The increased primary temperature causes the nuclear power to decrease because of the negative moderator effect. In few seconds reactor scram is triggered by the low loop flow rate signal and the control and shutdown rods will start inserting, the reactor power decreases along with the primary pressure and temperature. As a consequence DNBR increases and the reactor is brought to a controlled state.
An inadvertent opening of the pressurizer pilot operated safety valve with a simultaneous full opening of the main steam relief valves has an effect on DNBR in the beginning of a loss-of-coolant accident; the primary pressure decreases and voiding increases. This has effect on the core temperature and DNBR, but the magnitude of the effect is not clear.

Many currently marketed LWR designs feature varying numbers of naturally circulating decay heat removal loops. Large diversity of design configurations is available. Some vendors connect the loops to the primary system; Isolation Condensers (IC) or Emergency Condensers (EC) of BWRs and Passive Residual Heat Removal Systems (PHRHS) in PWRs. Some connect to the secondary side; Passive Heat Removal System – Steam Generators (PHRS-SG) in VVERs. Many vendors connect loops even to the containment; Containment Cooling Condensers (CCC), Passive Containment Cooling Systems (PCCS), (both in BWRs) and Passive Heat Removal System – Containment (PHRS-C) in VVER.

Design details, e.g. the exact geometry of the heat exchangers, vary a lot between different vendors. Most designs rely on a water pool outside the containment as the heat sink, however. Moreover, the general features of naturally circulating loops are similar in all designs. These safety systems are designed to operate without an external power source and relying on relatively small gravitational pressure differences. The vendors are in the possession of design-specific performance data for the systems, but the coverage of the data is not widely known. In particular, it is unclear as to what extent the vendor testing covers phenomena and inherent failure mechanisms that could prevent or hamper the intended operation of the system. It is particularly noteworthy that the containment cooling loops often operate at low (~atmospheric) pressure, meaning that they are susceptible to boiling oscillations due to large water-steam density difference. While oscillating flow may be an efficient heat removal mechanism, it causes dynamic loads and consequent fatigue on system piping, containment penetrations, pipe supports, and associated vessels.

The PHRS-C system of AES-2006 design was selected to be studied experimentally. In the design of the selected passive system facility also the possibility to study the effects of aerosols on the outer surface of the heat exchange tubes are taken into account. During an accident, aerosols will deposit preferably by gravity and thermophoresis, both of which tend to drive the aerosols towards the containment heat exchangers. Jorma Jokiniemi from University of Eastern Finland (UEF) will assist in the design of the heat exchanger system to enable later addition of aerosol injection and aerosol detectors. Aerosol tests will be performed later as a subject to separate funding.

1.2 Objectives and expected results

The main aim of the project is to ensure the operation of safety related systems or the efficiency of the procedures in accident and transient situations of nuclear power plants. An integral test facility, such as PWR PACTEL, offers a good possibility to carry out tests which supplement test campaigns in the other facilities (PKL) or make independent tests to study phenomena relevant to the safety of nuclear power plants (pump trip etc.) As a result, counterpart-like tests give information of parameter effects such as a smaller scaling ratio or a higher pressure level (PWR PACTEL/PKL) when certain operator actions or system activation set points are used.

The limiting factors of the operation of passive heat removal loops were surveyed and the PHRS-C system of AES-2006 design was selected in 2015 to be studied experimentally. The ultimate goal is to identify physical mechanisms that can reduce performance or prevent the functioning of the loop, to help recognizing conditions in which the functioning of the system could be endangered and to suggest ways assuring the operation. The design work of the facility began in 2016.

The expected results of the INTEGRA project are:
- LUT participates in the OECD/NEA PKL Phase 4 project with the PWR PACTEL facility,
- The PWR PACTEL experiments to study the flow reversal due to a pump trip,
- A system to investigate the fundamentals of the PHRS-C passive system of AES-2006 design
1.3 Exploitation of the results

The PWR PACTEL tests in the OECD/NEA PKL Phase 4 project are analysed together with the results from the other test facilities by the organisations participating the project. Extensive computational efforts are used in the OECD/NEA PKL Phase 4 project to analyse the transients ran in the test facilities and to transfer the data to the reactor scale. LUT will participate the OECD/NEA PKL Phase 4 project with two PWR PACTEL experiments.

The results of the tests on flow reversal due to a pump trip are used to ensure the computational capabilities in predicting the phenomena in plant analyses. The results are available for the analyses in the second half of 2018.

The survey of the factors limiting the operation of passive heat removal circuits were used to choose the passive system to be constructed. The ultimate goal is to help the system designers to avoid the situations where the selected PHRS-C passive heat removal system of AES-2006 design would not operate as planned. The results of the studies can be fully exploited after 2018.

1.4 Appropriateness of the project to SAFIR2018 programme

The proposed INTEGRA project is a part of national nuclear safety research that develops and creates expertise, test facilities and computational methods to be used to solve nuclear safety related problems. It has significant international connection through the OECD/NEA projects. The INTEGRA project provides information that helps both the regulator and the licensees assess where the biggest uncertainties are in the performance of naturally circulating and nitrogen driven systems. The project is also a solid part of the national research infrastructure maintaining and development of which was pointed out in the recommendations by the international evaluators of SAFIR2014 programme.

1.5 Education of experts

The researchers working in this project gain expertise in designing and constructing the test environment for the studies of various thermal hydraulic problems as well as in analysing the problems with computational methods. A significant portion of the work will be carried out by master’s thesis workers. In the long term this kind of work leads to doctoral theses as well. In addition to the bachelor’s and master’s theses one dissertation is expected in the project.
2. Work plan

The INTEGRA project consists of thermal hydraulic tests (planning, conducting, analysing, and reporting) with the PWR PACTEL integral test facility. The planned PWR PACTEL tests in WP1 are chosen from a review of the transients important for reactor safety. In 2016 the effects of nitrogen from an accumulator tests were carried out. The tests continues in 2017 with the flow reversal due to a pump trip tests. The inadvertent opening of SV and MSRT tests have to be cancelled due to financial cuts. In WP2 fundamental of the PHRS-C passive heat removal circuit of AES-2006 design is investigated to reveal undesired features of the system that may endanger the functioning of the system. In 2017 the selected passive system will be constructed. Characterizing tests will be in 2018.

Figure 1. The overall schedule of INTEGRA for 2015-2018.

<table>
<thead>
<tr>
<th>INTEGRA, Work packages and Tasks</th>
<th>2015</th>
<th>2016</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
<tbody>
<tr>
<td>WP1 - Integral tests with PWR PACTEL</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T1.1 Participating in the OECD/NEA PKL Phase 4 project</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T1.2 Effects of nitrogen from an accumulator</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T1.3 Flow reversal due to a pump trip</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T1.4 Inadvertent opening of SV and MSRT</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WP2 - Passive heat removal circuits</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T2.1 Fundamentals of passive systems</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T2.2 Designing and construction of test ...</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T2.3 Testing of selected systems</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WP3 - Project management</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T3.1 Project management</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T3.2 SAFIR2018 midterm seminar</td>
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<td></td>
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<tr>
<td>T3.3 Publications</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T3.4 OECD PKL project participation fees</td>
<td></td>
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<td></td>
</tr>
</tbody>
</table>

2.1 Integral tests with PWR PACTEL (WP1)

The Work Package contains participating in the OECD/NEA PKL Phase 4 project and the experiments studying the flow reversal due to a pump trip.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>6.5</td>
</tr>
</tbody>
</table>

2.1.1 Participating in the OECD/NEA PKL Phase 4 project (T1.1)

LUT has been participating in the OECD/NEA PKL Phase 3 project with the PWR PACTEL experiments and will also participate the OECD/NEA PKL Phase 4 project with two PWR PACTEL experiments (impact of nitrogen on cool-down/heat removal). The PWR PACTEL tests complements the PKL experiment(s) on IB/SB-LOCA. The experiments will be planned in co-operation with the project partners in 2017. The experiments will be carried out in 2018 and/or 2019.

The participation fee of Finland in the OECD/NEA PKL Phase 4 project (60 k€) goes via LUT. The fee for 2017 is 20 k€.

2.1.2 Effects of nitrogen from an accumulator (T1.2)

Finalized in 2016.
2.1.3 Flow reversal due to a pump trip (T1.3)

A loss of one reactor coolant pump causes the motor torque in one primary pump to be nullified, resulting in a decrease of the reactor coolant pump’s speed and reactor coolant flow rate in the affected loop. Yet, the energy stored in the fuel pellets continues to be transferred to the coolant whose flow rate decreases which causes a primary coolant temperature and pressure rise. The DNBR in the core decreases which could cause the outbreak of DNB crisis.

A flow reversal occurs in the affected loop due to reversal of pressure distribution in this loop caused by the other running pumps. The final flow conditions are characterized by a slight overflow in the intact loops and a backflow in the loop with the idle reactor coolant pump. The reactor is stabilized at shutdown with a reduced core coolant flow.

With the PWR PACTEL experiments the asymmetric flow in the loops and in the core as well as the core power behaviour and control are studied. In these plans we are prepared to make two experiments. The APROS and/or TRACE codes will be used to support the experiment planning. The same codes will be used in the post-test calculations to improve the simulation models. More detailed plans will be finalized based on the pre-test calculations and the experiments will be carried out in 2017. The test results will be analysed and reported in 2018. Also post-test calculations with the APROS and/or TRACE codes will be carried out and reported in 2018.

The distribution of the required resources is ~20 % for the experiment planning including pre-test calculations, ~30 % for the experiments, ~30 % for the analyses of the experiment results and reporting, and ~20 % for the post-test calculations and reporting.

2.1.4 Inadvertent opening of SV and MSRT (T1.4)

An inadvertent opening of the pressurizer pilot operated safety valve with a simultaneous full opening of the main steam relief valves has an effect on DNBR in the beginning of a loss-of-coolant accident; the primary pressure decreases and voiding increases. This has effect on the core temperature and DNBR, but the magnitude of the effect is not clear.

Due to financial cuts this task had to be cancelled. Also plant scale simulations with a general model in the COVA project by VTT has to be cancelled. So, no activity in the SAFIR2018 Research Programme.

2.2 Passive heat removal circuits (WP2)

LUT investigates the fundamentals of the PHRS-C passive system of AES-2006 design in order to observe and detect the physical phenomena which could prevent the system to function as designed. Studies on the effect of aerosols are also considered in the design of the facility with help of Jorma Jokiniemi from UEF.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>6</td>
</tr>
</tbody>
</table>

2.2.1 Fundamentals of passive systems (T2.1)

Finalized in 2015.

2.2.2 Designing and construction of test environment of selected passive system (T2.2)

The selected system is designed in 2016 and will be constructed in 2017. Also the system testing will begin in 2017. Characterizing tests will be in 2018. Aerosol tests are scheduled after 2018. The costs of Jorma Jokiniemi’s work in the planning of the aerosol measurement systems is included in the external services in the task.
2.3 Project management (WP3)

Participation and preparation of SAFIR meetings and other project management related work and costs. Participating to the SAFIR2018 midterm seminar and writing a journal article.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>3</td>
</tr>
</tbody>
</table>

2.3.1 Project management (T3.1)

Participation and preparation of SAFIR2018 reference group meetings.

2.3.2 SAFIR2018 midterm seminar (T3.2)

Participation of the SAFIR2018 midterm seminar.

2.3.3 Publications (T3.3)

Writing a journal article.

2.3.4 OECD/NEA PKL Phase 4 project participation fees (T3.4)

The OECD/NEA PKL Phase 4 project participation fee.
## 3. Deliverables and milestones 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>First plan of the PWR PACTEL experiments in the OECD/NEA PKL Phase 4 project</td>
<td>2.5</td>
<td>1.12.2017</td>
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<tr>
<td>D1.3.1</td>
<td>Research report on the PWR PACTEL tests studying flow reversal due to a pump trip</td>
<td>4</td>
<td>30.6.2018</td>
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<tr>
<td>N/A</td>
<td>Milestone: Flow reversal due to a pump trip experiment plans ready.</td>
<td>N/A</td>
<td>30.6.2017</td>
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<tr>
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<td>Criterion for approval: Experiment plans ready.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: Flow reversal due to a pump trip experiments done.</td>
<td>N/A</td>
<td>31.12.2017</td>
</tr>
<tr>
<td></td>
<td>Criterion for approval: Experiments successfully done.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: Flow reversal due to a pump trip experiment simulation reporting ready.</td>
<td>N/A</td>
<td>30.6.2018</td>
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<tr>
<td></td>
<td>Criterion for approval: Simulation report ready.</td>
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<tr>
<td>D2.2.1</td>
<td>Facility description of the system design to investigate the fundamentals of the PHRS-C passive system of AES-2006 design</td>
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<tr>
<td>N/A</td>
<td>Milestone: Construction work of the passive system began.</td>
<td>N/A</td>
<td>30.4.2017</td>
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<tr>
<td></td>
<td>Criterion for approval: Construction work started.</td>
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</tr>
<tr>
<td>D3.3.1</td>
<td>A journal article on relevant topic of the INTEGRA project</td>
<td>3</td>
<td>31.12.2017</td>
</tr>
</tbody>
</table>

| Total pm           | 15.5                                              |
4. Project organisation

Nuclear engineering forms the project organisation at LUT. LUT is responsible for the whole project. Mr. Vesa Riikonen will act as the project manager. This project is planned to be carried out by LUT.

Since this work at LUT is mostly dealing with experiments and constructing test facilities, it is impossible beforehand to decide the exact working hours of a single person. Thus, only an estimated person months is presented with a full list of the persons who can be involved.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vesa Riikonen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP2, WP3</td>
<td>5</td>
</tr>
<tr>
<td>Markku Puustinen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Antti Räsänen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>1</td>
</tr>
<tr>
<td>Heikki Purhonen</td>
<td>Research director</td>
<td>LUT</td>
<td>WP1, WP2, WP3</td>
<td>0,5</td>
</tr>
<tr>
<td>Virpi Kouhia</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>2,5</td>
</tr>
<tr>
<td>Jani Laine</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Joonas Telkkä</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>1,5</td>
</tr>
<tr>
<td>Lauri Pyy</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP2</td>
<td></td>
</tr>
<tr>
<td>Harri Partanen</td>
<td>Design engineer</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>0,5</td>
</tr>
<tr>
<td>Eetu Kotro</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>1</td>
</tr>
<tr>
<td>Kimmo Tiilinen</td>
<td>Research Trainee</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Ilkka Saure</td>
<td>Technician</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>1</td>
</tr>
<tr>
<td>Vesa Tanskanen</td>
<td>Post-doctoral researcher</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Juhani Vihavainen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Otso-Pekka Kauppinen</td>
<td>Doctoral student</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>2,5</td>
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<tr>
<td>Ville Rintala</td>
<td>Doctoral student</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Heikki Suikkanen</td>
<td>Post-doctoral researcher</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Juhani Hyvärinen</td>
<td>Professor</td>
<td>LUT</td>
<td>WP1, WP2, WP3</td>
<td></td>
</tr>
<tr>
<td>Jorma Jokiniemi</td>
<td>Professor</td>
<td>UEF</td>
<td>WP2</td>
<td>1 *</td>
</tr>
<tr>
<td>N.N.</td>
<td>Research Trainee</td>
<td>LUT</td>
<td>WP1 +</td>
<td></td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>15,5</strong></td>
</tr>
</tbody>
</table>

* Work included in the external services, no effect on person months of LUT
5. Risk management

The risks in the project are associated with the unexpected malfunctioning of research equipment and potentially limited availability of human resources, materials, and laboratory work.

Failures to complete tests due to hardware or other systemic issues can take a long time that may lead in significant delays for the planned activities. These risks can be minimized by careful operation of the measurement devices, ensuring that the personnel operating the devices have sufficient knowledge how to operate them, and the regular maintenance and renewal of the research equipment.

The work proposed here does require certain competences that only the senior research staff possesses. In case of unavailability of the necessary knowledge, project scope may need to be reduced and/or emphasis shifted to areas which can be completed using the available resources.

There are new users of electric power in LUT laboratories. It means that scheduling the experiments is now more challenging than before.

The foregoing assumes that catastrophic events such as massive fire in the laboratory or LUT campus can be excluded from the risk assessment.
References

/1/ Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, PWR PACTEL nitrogen experiments, Research report INTEGRA 1/2016, Lappeenranta University of Technology / Nuclear Engineering, Lappeenranta, 2016, 23 + 17 s.


/7/ Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, Cool down under Natural Circulation Conditions in Presence of Secondary side Isolated Steam Generators, Research Report, PAX 1/2013, Lappeenranta University of Technology / Nuclear Safety Research Unit, Lappeenranta, 2013, 17 + 3 s.

/8/ Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, Station blackout experiments, Research Report, PAX 1/2014, Lappeenranta University of Technology / Nuclear Safety Research Unit, Lappeenranta, 2014, 20 + 18 s.

/9/ Vesa Riikonen, Virpi Kouhia, Otso-Pekka Kauppinen, PWR PACTEL flow reversal experiments, Research Report, INTEGRA 1/2015, Lappeenranta University of Technology / Nuclear Engineering, Lappeenranta, 2015, 10 + 3 s.
### Integral and separate effects test on thermal-hydraulic problems in reactors

#### Expenses

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat &amp; supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Memb fee</th>
<th>Other</th>
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<th>Fennovoima</th>
<th>Fortum</th>
<th>TVO</th>
<th>Aalto</th>
<th>LUT</th>
<th>VTT</th>
<th>NKS</th>
<th>Areva</th>
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</thead>
<tbody>
<tr>
<td>WP1 - Integral tests with PWR PACTEL</td>
<td>6,5</td>
<td>80,0</td>
<td>5,0</td>
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#### Comments:
- T2.2 Jorma Jokiniemi’s planned work of 1 person month is included in the external services.
- T3.4 OECD/NEA PKL Phase 4 participation fee 2017 20 k€ (2018 20 k€, 2019 20 k€).
KATVE
Kriittisyys- ja
Turvallisuusanalyysi
Valmiuksien
Kehittäminen
Nuclear Criticality and Safety Analyses
Preparedness at VTT

Pauli Juutilainen, Asko Arkoma, Eric Dorval, Silja Hääkinen,
Toni Kaltiaisenaho, Petri Kotiluoto, Jaakko Leppänen, Timo
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VTT Technical Research Centre of Finland
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1. Research theme and motivation

The main goal of the KATVE project is to maintain and develop competence in performing various safety analyses for the authority and the utilities. The topics of the KATVE project are mainly related to radiation shielding, criticality safety, reactor physics, activation analysis and reactor dosimetry. As a somewhat separate task, a multidisciplinary in-depth analysis, including heat transfer and fuel performance analyses, is also carried out within the project for a spent fuel dry storage cask.

The volume of actual radiation shielding research at VTT has been very small during the past couple of years, mainly since the necessary analyses have been performed by experienced personnel using established international calculation codes. However, as the previous experts have retired and as the calculation methods are constantly developing, radiation shielding is again a relevant research topic. Making shielding calculations with general-purpose Monte Carlo transport programs like MCNP [1] or Serpent [2] requires special techniques, since the statistics become very poor in shielded regions with low particle densities when using standard analog Monte Carlo simulation. On the other hand, deterministic tools dedicated for shielding calculations are used only rarely and, hence, their use, properties and limitations need to be examined separately. Consequently, effort is needed to revive the capability to perform radiation shielding analyses for neutrons, photons and beta particles with associated secondary radiation.

Reactor physics research is done also in other SAFIR2018 projects, e.g., MONSOON (generation of group constants for nodal codes) and SADE (reactor dynamics). The main emphasis in the reactor physics research of KATVE is in the generation of source terms, in practice nuclide inventories, heat sources and gamma sources, for various analyses involving spent nuclear fuel. Such source terms are every now and then requested by the utilities and the authority, and currently the expertise in the generation of such terms is concentrated on few key personnel. In addition to expertise, credible generation of such source terms requires proper validation of the calculation systems, which is also done in the framework of the KATVE project.

Also the competence in criticality safety has somewhat decreased during the past couple of years because of leaving experts. General criticality safety studies have been performed already in SAFIR2014 program. In the current project, the general criticality safety research continues by familiarizing new experts with domestic and international practices and regulations, in addition to which the knowledge is deepened by concentrating on a more specific topic, burnup credit (BUC).

Activation analysis and reactor dosimetry are both related to studying the neutron flux and sometimes also gamma flux at the periphery of a reactor. The output quantities are, however, somewhat different: in activation analyses the activation of structural materials is of interest, with emphasis on long-lived nuclides, whereas reactor dosimetry is all about determining fast neutron fluences and displacements-per-atom (dpa) for reactor pressure materials in surveillance positions and at the reactor pressure vessel. Since typical nodal codes are not applicable to calculations outside the reactor core, the calculations must be done using special techniques and tools. The calculation system currently used at VTT for this kind of analyses is cumbersome, outdated and no longer supported by the developers, and therefore the current code system should be hastily replaced with more modern calculation tools. Since the current responsible in reactor dosimetry will leave VTT by the end of 2016, at least one new expert should also be familiarized with the field, involving for instance spectrum unfolding and gamma spectrometry on top of more traditional reactor physics techniques.

Because of overlapping project organisations, KATVE project is connected to the SAFIR2018/MONSOON project. Serpent code development is done in both of the projects, but with very different focuses.

1.1 Background and state-of-the-art

In criticality safety, the standard level, or norm, is to have an appropriately validated and evaluated calculation system. Having the possibility to take burnup credit into account in the calculations is an improvement to the standard level. When applying burnup credit in spent fuel criticality safety analyses, the depletion calculations need to be properly validated. Traditionally, the depletion calculation is validated against measured nuclide inventory data.
Another way to perform the validation is a combined criticality and depletion validation, but unfortunately there exists very little public data on such combined experiments.

VTT’s abilities to perform criticality safety analyses lag somewhat behind this international level. In addition, the know-how has been jeopardised as experts in the field have left or retired. To some extent this gap has been filled up already in the SAFIR2014 programme. The proposed project will bring the tools and knowledge back on track and to the standard level. Once this is done, work to advance them to the state-of-the-art can be considered.

The Serpent Monte Carlo code [2] has been developed at VTT since 2004, and it has an international user community of approximately 560 users in 162 organizations around the world. In the KATVE project, Serpent is used as a calculation tool for criticality safety and radiation shielding as well as producing the source terms, i.e. nuclide inventories and decay heat, for heat transfer analyses. The work involves taking advantage of some of the unique and most recent features in the Serpent code, including unstructured mesh-based geometry types, which can be used for modelling complicated storage geometries and passing decay heat distributions to CFD calculations without intermediate steps or loss of modelling resolution.

The development of a photon transport mode in Serpent was started in the KÄÄRME project of SAFIR2014 as a M.Sc. thesis [3]. The work is continued in the KATVE project, aiming to develop Serpent into a versatile state-of-the-art tool for radiation shielding calculations. This is achieved by further developing the photon interaction physics models, by including all relevant photon generation mechanisms, including Bremsstrahlung and spontaneous fissions, into the radioactive decay source and by developing general-purpose, easy to use variance reduction techniques.

Both in activation analysis and in neutron dosimetry there are needs to update and modernise the calculation system. So far, activation analyses have been performed with the discrete ordinates code package DOORS [4] which is no longer supported or actively developed. This has resulted in serious convergence and memory allocation problems on current calculation clusters. Internationally, continuous-energy and multi-group Monte Carlo methods are becoming more and more popular in the solution of activation analysis problems, and hence the possibility to introduce such methods at VTT should be examined. It should also be ensured that the new calculation systems, when taken into use, provide reliable results VTT’s typical calculation cases. Monte Carlo programs often require less user interaction than the deterministic codes, and hence the overall time spent in the analyses may actually decrease when making the switch, even though the CPU time requirements of Monte Carlo codes are typically larger by at least one order magnitude.

The neutron irradiation affects the structural integrity of many critical components within nuclear reactors. Neutron dosimetry is used to estimate the neutron doses of these components by combining measurement data with computational results. This requires similar calculation capabilities as activation analysis, but in addition gamma spectrometry and neutron flux spectrum unfolding or, in other words, spectrum adjustment based on experimental data should be mastered. This can only be achieved with sufficient expertise in dedicated computer programs. In 2013, the competence of VTT in this field was significantly reduced along with the retirement of a well-established expert, and the competence is again at stake in 2017, since the current responsible leaves VTT at the end of 2016.

In this project CFD and heat transfer simulations have a supporting role. The main objective is to demonstrate a calculation chain, in which the temperature distributions from a CFD calculation are used as boundary conditions in a fuel integrity study. The heat source to the CFD calculations is obtained from a reactor physics calculation. The CFD calculations are made using state-of-the-art CFD calculation tools. The different material properties of gas and solid may, however, pose a challenge to the numerical behaviour of the simulation.

From the cladding point of view the most important issues to be considered during dry storage are the cladding creep and the behaviour of hydrides in the cladding. They may result in creep rupture or hydride induced failures. Hydrides also make the cladding more brittle and thus complicate the fuel handling. Of these two failure modes, creep rupture is considered to be dominant. The fuel integrity analyses in KATVE will be performed using a recently developed version of ENIGMA fuel performance code [5]. Literature surveys were made in SAFIR2014, and the creep correlations found during that work were implemented into the code. Consequently, the new version is applicable to fuel modelling in dry storage conditions.

The capability to analyse dry storage is relevant since it has been considered as a possible option also in Finland. Therefore, national competence for the assessment of the safety of dry cask storage facilities should be developed. Internationally, similar studies have been made for example at CIEMAT by using the ANSYS Fluent CFD code and FRAPCON-3.4 fuel performance code [6].

1.2 Objectives and expected results

The objective of the project is to improve the preparedness to perform safety analyses in the field close to reactor physics as well as increase the knowledge of the current researchers and educate new experts to the field. The
main analyses are in the field of criticality safety and radiation shielding. Additional analyses are activation of structural materials and neutron dosimetry. Moreover, source terms are often needed for analyses also outside the scope of this project. These source terms comprise nuclide inventories and decay heat. In this area, the ability to build a complete analyses chain from the heat source through the CFD calculation for heat transfer to the integrity of fuel is an important outcome.

The objective of the criticality safety work is to have an appropriately validated calculation system that has the possibility to take into account also the burnup credit. In addition to the tools, also the knowledge on how to perform a criticality safety analysis is revived. In the four years of the programme, the criticality validation package should be finalised and depletion validation of the calculation tools should be in good progress. The validation package and burnup validation reports are concrete results of the project. In addition, the calculation tools are developed to take into account the burnup credit in an easy and manageable way.

Serpent is developed into a practical simulation tool for criticality safety, radiation shielding and other applications involved in the project. Once the work is completed, the code is capable of performing reliable criticality safety analyses, producing source terms for radioactive inventory and decay heat calculations, as well as calculating neutron and gamma dose rates in complex geometries using the best available knowledge on interaction physics and state-of-the-art variance reduction techniques. The use of the continuous-energy Monte Carlo method with unstructured mesh-based geometry types and CFD code coupling constitute a one-of-a-kind calculation scheme for the safety analysis of spent fuel storage facilities.

Shielding analyses involve use of lighter codes like Microshield [7], in addition to the heavy Monte Carlo calculation. An important objective of this project is to learn to use the code with the source terms calculated with Serpent and ORIGEN-S [8].

Neutron transport calculations involving strong shielding are necessary for providing initial data for activation analyses and neutron dosimetry. The current calculation codes based on discrete ordinates method are outdated and the aim is to replace them with more modern, actively supported codes. This requires both a survey of VTT’s activation analysis needs and training on the new codes. The primary objective also in reactor dosimetry is to maintain the previous competence by familiarizing new experts with the field.

The main objective of the in-depth study on dry storage cask is to demonstrate functioning of the whole calculation chain, which includes 3D heat source calculation with Serpent, CFD analysis for heat transfer and fuel integrity analysis corresponding to different cooling times and, thus, temperature profiles for the fuel cladding. Capability to perform this kind of complicated analyses in-house is considered a significant achievement.

### 1.3 Exploitation of the results

The project develops tools for and knowledge on performing safety analyses. With these tools VTT is better prepared for possible future assignments in the field. When the project is finished, the tools and methods should be ready, and the know-how on how the analyses in the scope of this project are to be made should be available and properly documented.

In criticality safety, the first version of the validation package will be available already after the first year although the development continues throughout the programme. Combining a depletion code and a criticality safety calculation code from scratch would require a lot of effort, but fortunately Serpent already has the built-in depletion capability which is readily usable. Hence, setting up the criticality calculation system around Serpent will be an efficient solution. This way the criticality safety development may also benefit from the contributions and experiences of Serpent user community.

As a versatile calculation tool, the Serpent code can also be applied for various safety-related tasks encountered in the storage and transport of spent nuclear fuel. At the end of the project, Serpent will be fully capable of generating complete gamma sources for irradiated materials including spent fuel, and performing physically valid gamma transport analyses in all kinds of geometries including problems with strong shielding. These new features of Serpent benefit both the domestic and international users of Serpent. The work also contributes to the education of new experts in the field and enhances VTT’s competence.

Little data is available on comparison between light and heavy calculation codes in radiation shielding. Such a comparison will benefit also the utilities as experience is obtained on how well the more simple methods work compared to heavier Monte Carlo calculations.

Nuclide inventory and decay heat calculations are used regularly to provide initial values for other analyses. The training of researchers to provide source terms already started in the SAFIR2014 programme, and source terms are also calculated in the current project using Serpent. The calculated source terms will be used in other parts of the project. Experiences in source term calculations also increase the preparedness to perform similar analyses for the utilities and the regulator by request.
Once the new tools are taken into use, the activation analysis system will be available in other projects. The modernisation of the system will ensure the capability to perform such analysis as the newer codes are more reliable and easier to use. Checking the usability in older cases by recalculating them ensures smooth transition to a new code system and also ensures capable staff. Neutron dosimetry is required on a yearly basis, in conjunction with the research on structural integrity performed at VTT. Consequently, the updated tools and knowledge are needed, and they will be readily useable once the work in this project is done.

Dry storage is widely used in the world for interim storage of spent fuel, and this is an option also in Finland. Internationally, the research on fuel behaviour in dry storage is active, and extending the capabilities of a fuel performance code to model the dry storage period provides a tool needed to participate in such international activities. Obtaining the cladding temperature from CFD calculations demonstrates a calculation chain that can be used in evaluation of dry cask storage systems or transport containers. The cooling of the dry cask is based on passive systems. However, it has to be ensured that the cladding temperature does not rise too much because the neutron absorbing materials in the cask are generally poor heat conductors.

1.4 Appropriateness of the project to SAFIR2018 programme

The purpose of the SAFIR2018 programme is to ensure that the authorities have enough knowledgeable staff and other resources to work out new issues regarding safe use of nuclear installations [9]. The aim of this project is to increase the preparedness of VTT to perform safety analyses in the field close to reactor physics. This is achieved by improving and developing tools used in these analyses and educating new experts in the field.

Criticality safety and radiation shielding are increasingly important issues as new facilities are planned, and the preparedness of VTT in these fields is not quite as high as it should be. Some further effort is required to reach an acceptable level with both the knowledge and the tools. The source terms, nuclide inventory and decay heat, are important not only for these analyses but also for many other studies not covered in this project.

1.5 Education of experts

The whole project is set up to improve the knowledge in the safety analyses chosen for this project. These fields need stronger competency at VTT, since a lot of it has been lost recently due the loss of personnel.

Developing new calculation methods in the Serpent code educates new experts in radiation shielding and other safety analyses related to the storage of spent nuclear fuel. The work also begins a new D.Sc. project (Kaltiainenaho) on extending the modelling capabilities of Serpent from reactor analysis to more general-purpose radiation and particle transport applications.

The project also provides possibilities to perform student assignments, Master’s theses or other theses in the fields chosen for the project. This would also educate new experts in the field.
2. Work plan

The main goal of the whole project is to ensure competent staff and adequate tools for safety analyses close to reactor physics. The actual safety analyses cover criticality safety and radiation shielding. In addition, work is done in activation analysis and neutron dosimetry. As source terms, nuclide inventory and decay heat have important roles in many safety analyses, the source term generation is also tackled by this project.

The work has been divided into four work packages according to the analysis under consideration. In addition, a work package has been set up for project management, QA issues and international co-operation. The contents of the work packages and their tasks are described in higher detail in the following. The main emphasis in the task descriptions is put on the year 2017, which is the third year of the four-year project.

The original project plan was drafted for a yearly budget of 384 k€, but the project budget was cut by about 40% such that the realized funding in 2015 was only 229 k€. In 2016, the funding decreased further to 178 k€. Due to the significant cuts in the project volume, many of the research activities included in the original plan had to be postponed or dropped. Code validation, gamma transport development, burnup credit and in-depth analysis of Dry Storage were considered high-priority topics when re-allocating the limited resources. The decrease of funding slightly continued for 2017 to 150 k€. On the other hand, some funding was granted for the new RADICAL project, but following the instructions of the SAFIR management board, it was merged to a new subtask for the KATVE project. Thus the final volume for the KATVE of 2017 is 200 k€. The funding cut was rather small compared to the applied sum, so the objectives listed in the original application should be mainly achievable.

2.1 Work package 1 (WP1): Criticality safety

The aim of this work package is to increase the know-how in criticality safety, and to develop tools and methods required in the analyses. The goal is that after the four-year programme VTT is capable of performing properly validated criticality safety analyses for the authority and the utilities.

The criticality knowledge at VTT has decreased lately. This is due to personnel leaving or retiring. Also the methods, especially the criticality validation, should be improved from the current state. Some of this work has been done in the previous SAFIR2014 programme. However, this work needs to be continued as is stated in the SAFIR2018 Framework Plan [9].

The main concern throughout the programme is the validation package [10]. The package will be brought to a level where it can be used as a proper tool in criticality validation. It will then be used to define the upper safety limit used for the analyses as stated in standards and other requirements.

The rather broad topic of burnup credit will be studied in the second task of the work package. First, a literature survey on the use of burnup credit and the associated Gd-credit both in Finland and abroad will be made. Later in the project, preparations for applying burnup credit at VTT will be made according to the suggestions of the survey. Using burnup credit requires at least validation of the depletion calculations, on top of which also the criticality safety calculations should be validated for systems involving spent fuel. The burnup credit studies and validations will be mainly performed with Serpent, which includes a built-in burnup calculation capability. The results are compared to other burnup calculation codes if considered beneficial.

Studying criticality safety analysis practices is easier with a practical application. Since data for most relevant systems, e.g. spent fuel pools, is confidential, it was decided to pick dry storage cask of spent fuel as the main system for criticality safety studies. Public data are available for certain dry storage cask designs, and using this case creates also synergy with WP4. The ultimate goal of the third task is a complete criticality safety in which also the needs for burnup credit are considered. This work is planned for the last year in the programme.

Person work months allocated to WP1 are given in the table.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>3.46</td>
</tr>
</tbody>
</table>
2.1.1 Task 1 (T1.1): Validation package

The validation package for criticality safety has been extended yearly by adding inputs from new critical experiments to both MCNP and Serpent. During 2016 VTT managed to obtain the MCNP inputs of over 2000 critical experiments that are used in the JEFF nuclear data project for validating the new JEFF cross section libraries with MCNP. These MCNP inputs can be converted to Serpent inputs and used for the criticality safety validation of Serpent. The cooperation with the JEFF-project will be deepened by participating in the validation of the JEFF-3.3 nuclear data libraries using Serpent.

During 2017, the work of converting the newly obtained 2000 critical experiments from MCNP inputs to Serpent inputs will be started by considering the most relevant (LWR/UOX) cases first. The conversion will be mostly automated using an existing conversion script, which will, however, have to be slightly updated for this task.

The building up of the MCNP critical experiment library will also continue in 2017 based on selected experiments from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [11]. For automatic trend analyses using the validation script, the values of several key parameters such as coolant boron concentration, lattice pitch and fuel enrichment have to be added to the inputs in machine readable form.

The validation script will be maintained through the year and updated if needed. One of the future updates that is needed is an easy way to switch certain simulation options such as the unresolved resonance treatment and Doppler Broadening Rejection Correction (DBRC) [12] on and off by demand by only modifying the input-file for the validation script.

2.1.2 Task 2 (T1.2): Burnup credit

A state-of-the-art report on the burnup credit (BUC) practices in Finland and abroad has been prepared in the KATVE project during 2015 and 2016. Based on this report, the logical next step in the introduction of BUC in the reactor calculation system of VTT is the validation of the burnup calculation capabilities of the lattice physics codes used at VTT. The validation can be done either against other calculation codes or, preferably, against experimental data.

Most of the publicly available measurement data on spent fuel compositions is available in SFCOMPO (Spent Fuel Isotopic Composition) database [13]. Also reference [14] includes accurate PIE (Post Irradiation Examination) data for samples taken from VVER fuel assemblies at intermediate burnups. This data has been widely used in burnup credit validation purposes for VVER reactors and, hence, it would act as a very good starting point for the burnup calculation validation in KATVE.

In 2017, the burnup calculation of Serpent and CASMO are validated against the data in Reference [13]. The coolant and power histories of the PIE samples are taken into account in the burnup calculations at a high accuracy. The ultimate goal of the validation is to determine isotope correction factors (ICF) and their uncertainties for the most important burnup credit nuclides. The uncertainties will be determined using established standard techniques.

The validation effort is expected to continue in 2018 with data from the SFCOMPO database. The burnup validation effort benefits also source term calculation in task T3.1.

2.1.3 Task 3 (T1.3): Criticality analysis

This task was set up for general criticality analyses and for studying the national and standards and requirements in the field of criticality safety. These kinds of activities are not planned for 2017.

2.2 Work package 2 (WP2): Radiation shielding

The intention of this work package is to improve the know-how of the researchers at VTT on radiation shielding analyses, and to develop tools for radiation shielding analysis. The importance of this area seems to be increasing as new nuclear installations are being built and planned, and on the other hand old reactors are to be decommissioned. For instance, implementing the final disposal of spent fuel requires numerous radiation shielding analyses for the handling, transport, and encapsulation of the fuel. With this work package we want to ensure competent staff and adequate tools for such studies.

Development of a photon transport mode for the Serpent Monte Carlo code started in the KÄÄRME project in SAFIR2014. The initial incentive for this work was in the calculation of gamma heat deposition in coupled multi-physics simulations, but the capability opens new possibilities also for radiation shielding applications. The devel-
Development work is continued in the first task of WP2 with emphasis on completing the gamma physics models, implementing nuclide composition based gamma source generation, and development of variance reduction techniques. The variance reduction techniques are extremely important when solving radiation shielding problems with Monte Carlo. The same techniques are also applicable to neutron transport, and hence the development benefits also activity analysis and neutron dosimetry tasks in WP3.

The second task will deal with the shielding analysis itself. In addition to the heavy computation with Monte Carlo codes also lighter codes will be used and a comparison between the two approaches will be performed. Dry storage cask for spent fuel is used as the application also for this analysis, for the sake of synergy.

Funding for a new one-year project (RADICAL) focused on radiation transport applications in severe accidents was applied in 2017. The funding was partially granted, but it was decided to merge the project with KATVE 2017. A new task (2.3) was established to cover the work originally planned for the RADICAL project.

Person months allocated to WP2 are given in the table.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>7.2</td>
</tr>
</tbody>
</table>

2.2.1 Task 1 (T2.1): Gamma transport in Serpent

Specific topics for 2017 include completing the radioactive decay source mode by adding neutrons and photons emitted in spontaneous fission and bremsstrahlung produced in beta decay. The omission of these mechanisms is the most likely cause for the underprediction of evaluated dose rates in previous calculations [15], and it is important that this deficiency is corrected. The methodology is validated by computational benchmarks, and if possible, comparing to experimental measurements performed at the reactor tank of the decommissioned FIR-1 reactor, where the fuel is still in place.

Development of variance reduction techniques started in 2016 is continued in 2017. The methodology currently relies on weight window meshes read in the MCNP WWINP format, together with a built-in light-weight importance solver based on the response-matrix method. The support for MCNP format mesh is extended to cover irregular mesh types and energy dependence is added in the built-in solver. More effort is also devoted to testing the methods in challenging shielded geometries.

The recent developments in the photon transport mode and variance reduction are described in two papers submitted to the M&C 2017 conference, and the participation is included in the travel costs of 2017.

Coupled neutron-photon simulation mode was implemented in 2016, and the work continues with photon-neutron coupling in 2017. In practice this implies including photonuclear reactions, in which neutrons are produced by high-energy photon interactions. This mode is considered important for the accurate modelling of detector responses in neutron dosimetry calculations. Neutrons produced in photo-fission may contribute significantly to NpO$_2$ and UO$_2$ fissile dosimeters and fission chambers in cases where the primary neutron flux is low but gamma flux is high, and high-energy photon interactions in structural materials produce additional neutrons that are picked up by the monitor.

2.2.2 Task 2 (T2.2): Shielding analysis of dry storage

The radioactive decay source mode and variance reduction techniques developed in Task T2.1 are put to practice in shielding calculations involving a spent fuel storage cask. This can be considered a realistic and challenging application for the developed methods. The source term is obtained from 3D assembly burnup calculations and photon dose rates are calculated on the on surface of the storage cask. The calculations are repeated for uranium oxide and MOX fuels, in order to test new source routines involving photons and neutrons from spontaneous fission. The calculations rely heavily on variance reduction, and can be considered a good test case especially for the built-in importance map solver. The results are compared to international regulations for the dose rates of dry storage casks.

2.2.3 Task 3 (T2.3): Radiation transport in severe accidents

This task includes work originally planned for the RADICAL project, with the purpose of addressing the computational challenges related to shielding and dose rate calculations in severe reactor accidents and other radiological
threats. The original plan was to cover the entire calculation sequence from source term formation to the evaluation of radiation dose rates, including the preparation of CAD-based geometry models and weight window meshes for variance reduction. The calculation sequence was to be set up in such way that it would enable rapid response to reactor accidents and other radiological threats, involving complex irregular geometry structures and source terms consisting of irradiated fuel with hundreds of radioactive isotopes. Since the budget was cut to less than 1/3 of that applied, most of these goals cannot be accomplished in KATVE 2017. By request of the SAFIR 2018 steering group, also all tasks related to validation and demonstration of the developed methodology were dropped.

The remaining tasks are related to source terms and transport physics relevant for accident scenarios. The difference to transport and storage conditions covered in other project tasks in KATVE 2017 is that the radioactive materials are not in solid form contained inside thick shielding, which usually allows making some simplifications. New source terms consisting of spontaneous fission neutrons and bremsstrahlung photons produced by locally deposited beta-radiation are implemented in Serpent 2. Development of physics routines involve improvements in the models applied to secondary electrons produced in photon interactions. Implementation of a rigorous explicit model for electron transport simulations cannot be considered a realistic goal. The work is instead focused on refining the existing models in which the secondary particles deposit their energy locally. Another important topic is the accurate evaluation of radiation doses from photons and electrons.

2.3 Work package 3 (WP3): Source terms and activation

The main aim of this work package is to ensure competent staff and adequate tools in the calculation of source terms for various analyses, activation analysis and neutron dosimetry.

The built-in burnup capability of Serpent provides the source terms for nuclide inventories, gamma source and decay heat. In the current project this capability is used to provide source terms for CFD analysis of dry storage casks.

The training of the staff in activation analysis started already during the SAFIR2014 programme. However, some of the codes used in the past have turned out to be inadequately documented and outdated. Consequently, the calculation system in this field needs updating.

Neutron dosimetry is included in this work package with a small contribution. The main purpose for this work is to keep the analysis system up-to-date including the dosimetry libraries. In addition, the competency of the staff is maintained and improved.

Person months allocated to WP3 are given in the table.

<table>
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<tbody>
<tr>
<td>VTT</td>
<td>0.9</td>
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</tbody>
</table>

2.3.1 Task 1 (T3.1): Source terms

This supportive task is dedicated to the calculation of source terms, for instance nuclide inventories and heat sources, for other tasks. New source terms are not required in 2017, so the task will be inactive.

2.3.2 Task 2 (T3.2): Activation analysis

This task is meant for activation analyses of reactor structures and nuclides, and renewal of the calculation system used at VTT. The task will be in active in 2017.

2.3.3 Task 3 (T3.3): Neutron dosimetry

Because the current responsible of reactor dosimetry leaves VTT in the end of 2016, the activities of task T3.3 in 2017 are dedicated to educating a new expert in the field. As a complete introduction to the calculation codes, auxiliary scripts and cross section libraries used at VTT, the neutron flux spectrum at the surveillance chain of Lovisa 1 reactor is adjusted using measured reaction rate data. The trial neutron flux spectrum is first calculated in BUGLE multi-group format using the Serpent model developed in Reference [16]. Then, the covariances in the calculated neutron flux are estimated using methods found in the ASTM standards [17] and covariances of the
measurements are estimated using the Matlab program developed in Reference [18]. Finally, the neutron flux is adjusted using, for instance, unfolding program LSL-M2 [19]. This study provides hands-on experience on all different tools required in reactor dosimetry. To preserve the confidentiality of the measured reaction rates, absolute values of the results are not included in the report.

2.4 Work package 4 (WP4): Heat transfer and fuel integrity

Investigations of heat transfer and fuel integrity in the long-term interim dry cask storage of spent nuclear fuel started in the SPEFU-project included in the SAFIR2014 programme. In the SPEFU-project, methods were developed and validated for the calculation of the heat transfer in a storage container with several fuel assemblies. When the heat transfer in several containers is calculated using Computational Fluid Dynamics (CFD), it is necessary to make a simplified model for fuel assemblies and cooling fins. Otherwise the computational work would be beyond present resources. The main result of the heat transfer calculation is the peak temperature of the cladding of the fuel rod. The peak temperature obtained from the CFD calculation is used in the creep modelling of the fuel rod cladding and analysing the fuel integrity in long-term storage.

The two parts should be combined in order to check the integrity of fuel cladding in realistic dry storage conditions. In the SPEFU-project these parts were not combined in lack of a common realistic case. The existing heat transfer experiments were suitable for modelling flow and heat transfer properties, but the cases were not relevant for checking creep properties due to too low temperature levels found in the experimental setups.

In WP4 of KATVE project, the full calculation chain aiming at fuel integrity analysis will be demonstrated. The analysis will be performed for a CASTOR type dry storage cask using realistic temperature levels for the spent fuel.

Person months allocated to WP4 are given in the table.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
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</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>2.15</td>
</tr>
</tbody>
</table>

2.4.1 Task 1 (T4.1) Heat transfer in dry storage cask

This task is dedicated to CFD analysis of dry storage cask. The heat transfer analyses have been performed already in 2015 and 2016 for different cooling times of the spent fuel, and hence this task will remain inactive in 2017 and 2018.

2.4.2 Task 2 (T4.2) Fuel integrity in dry storage cask

In 2017, the time evolution of the peak cladding temperature obtained from the CFD simulations made in 2016 (T4.1) will be used in the integrity analysis of the fuel rods in a CASTOR type dry storage cask. The analysis will be done using ENIGMA fuel performance code. The main parameter of interest is the cladding hoop strain, which will be studied with different power histories and diffusion constants. The hoop strain should remain below 1 % in dry storage conditions.

This simulation will be the last link in the demonstration of the calculation chain involving also heat source calculation with Serpent and CFD heat transfer calculation with OpenFoam. All phases in the calculation chain will be documented in an international publication.

2.5 Work package 5 (WP5): International co-op, documentation and management

This work package covers work related to project administration and management. In addition, it ensures resources to smaller activities related to the project but not necessary directly to specific work packages of the project. Issues like this are documentation needs for codes in or around reactor physics, writing papers to refereed journals or conferences on work done previously and where the subject of the paper relates to the topics of this project. An important part is also the international co-operation within e.g. OECD/NEA and the AER community. This co-operation will also be part of this work package.

Person months allocated to WP5 are given in the table.
2.5.1 Task 1 (T5.1): International co-operation

This task covers the international co-operation outside the direct scope of the previous work packages. It aims at taking care of Finland's obligations in NEA as officially nominated VTT experts. Participation in the NEA working groups and benchmarks is one of the most important ways of validating the methods and codes used in safety analyses. They also help in increasing the knowledge in the field.

The project is represented in OECD/NEA Nuclear Science Committee (NSC) as well as its “Working Party on Nuclear Criticality Safety”. The project will also participate in the activities of Expert Group of Assay Data of Spent Nuclear Fuel (EGADSNF) and the new expert group on used nuclear fuel (EGUNG). In 2017, a new representative will be nominated to the AER E working group, which covers the spent fuel, radwaste and decommissioning issues of VVER reactors. In addition, one project member has been recently nominated to NEA/MBDAV (Management Board for the Development, Application and Validation), which coordinates the JEFF project.

2.5.2 Task 2 (T5.2): QA, Documentation and scientific publication

This task was set up to cover small activities in code updating and documentation outside the scope of the other work packages in this project. Such activities are not planned for 2017.

2.5.3 Task 3 (T5.3): Project management

The task includes making plans for and supervising the project, collecting progress reports, arranging meetings and information exchange for the project’s reference group, possible ad hoc groups, etc.
3. Deliverables and milestones 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.3</td>
<td>Report or publication on the status of the validation package and validation results for LWR/UOX cases</td>
<td>1.96</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>N/A (T1.2)</td>
<td>Milestone: Burnup calculations performed for either Serpent or CASMO</td>
<td>N/A</td>
<td>30.9.2017</td>
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<tr>
<td>D1.2.2</td>
<td>Report or publication on the validation of burnup calculations and calculated ICFs</td>
<td>1.5</td>
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<tr>
<td>D2.1.5</td>
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<td>30.6.2017</td>
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<tr>
<td>D2.3.2</td>
<td>Conference paper or scientific journal article describing the current status of photon physics model in Serpent</td>
<td>1</td>
<td>31.5.2017</td>
</tr>
<tr>
<td>N/A (T2.3)</td>
<td>Milestone: Selection of the best course of action for the handling of electron and positron interactions in the photon physics routines (bremsstrahlung from beta decay and secondary photons)</td>
<td></td>
<td>31.5.2017</td>
</tr>
<tr>
<td>D2.3.3</td>
<td>Conference paper or scientific journal article on the applicability of the physics model used in Serpent for the purpose of radiation dose rate calculations</td>
<td>2.4</td>
<td>31.12.2017</td>
</tr>
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<td>1</td>
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<td>D4.2.1</td>
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<td>D5.1.1</td>
<td>Travel reports on international working group meetings participated</td>
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## 4. Project organisation

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pauli Juutilainen</td>
<td>Project manager</td>
<td>VTT</td>
<td>1.1, 1.2, 5.1, 5.3</td>
<td>3.1</td>
</tr>
<tr>
<td>Asko Arkoma</td>
<td>Research scientist</td>
<td>VTT</td>
<td>4.2</td>
<td>1.9</td>
</tr>
<tr>
<td>Eric Dorval</td>
<td>Research scientist</td>
<td>VTT</td>
<td>3.3</td>
<td>0.9</td>
</tr>
<tr>
<td>Silja Häkkinen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>1.2, 2.2</td>
<td>1.75</td>
</tr>
<tr>
<td>Toni Kaltiaisnaho</td>
<td>Research scientist</td>
<td>VTT</td>
<td>2.1, 2.3</td>
<td>5.3</td>
</tr>
<tr>
<td>Petri Kotiluoto</td>
<td>Research Team Leader</td>
<td>VTT</td>
<td>5.1</td>
<td>0.2</td>
</tr>
<tr>
<td>Jaakko Leppänen</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>2.1, 5.1</td>
<td>1</td>
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<tr>
<td>Timo Pätkikangas</td>
<td>Principal scientist</td>
<td>VTT</td>
<td>4.2</td>
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<tr>
<td>Riku Tuominen</td>
<td>Research scientist</td>
<td>VTT</td>
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<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>15.4</strong></td>
</tr>
</tbody>
</table>
5. Risk management

The main risk seen is the loss of personnel. In some of the tasks the know-how is concentrated on only one key person. Such areas have been identified and effort is made in this project to increase the number of persons who are able to perform such work. In case experts are lost from the project team, the need to educate new researchers is even stronger. This will naturally delay the progress of the project.

There is only little experience in using the newly implemented importance map solver of Serpent. It may turn out that the solver, in its current development phase, is inapplicable to the shielding calculations planned for 2017, task T2.2. If this is the case, the importance maps can be calculated using external tools like MAVRIC and imported to Serpent in WWINP format.

If significant differences to measured inventories are encountered in the depletion validation of Serpent, figuring out the causes for the differences and fixing possible errors in Serpent or the data files may require much more effort than included in the current plan. However, this risk should be quite low, since some depletion validation has been performed also in the past.

Some of the international meetings may be scheduled for the Finnish summer holiday season or other national holidays, and the research personnel cannot be expected to interrupt their holidays because of such meetings. This kind of overlaps are hard to predict and usually impossible to avoid. If some international meetings are missed, the interest group is kept up-to-date by reading thoroughly the minutes of the meetings and participating actively in the related discussions.
References


<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
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<td></td>
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<td>T3.1 Source Terms</td>
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<td>T3.2 Activation Analysis</td>
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<td>WP4 - Heat Transfer and Fuel Integrity</td>
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Comments:

Compulsory research facility costs from using the computation clusters of VTT (5 €/h) constitute most of Other costs.

Task T2.3 was originally applied as the separate RADICAL project that was granted 35 k€ by VYR, but merged with KATVE
SAFIR2018 Project plan

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

Jaakko Leppänen, Ville Valtavirta, Ville Sahlberg, Antti Rintala
VTT Technical Research Centre of Finland
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   1.3 Exploitation of the results .......................................................................................................................... 6
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1. Research theme and motivation

Nuclear reactor safety analyses involving coupled full-scale fuel cycle and transient calculations currently rely on a two-stage calculation scheme, in which the neutron transport physics at the fuel assembly level is first reduced into a set of representative group constants, which are then used as the input data for a simplified steady-state or dynamic full-core calculation. Group constant generation involves a procedure called spatial homogenization, which essentially implies the solution of the heterogeneous transport problem at the local (assembly) level. The procedure is repeated for different assembly types, burnups and reactor operating conditions, and the result is a complete data library, providing the sufficient building blocks for the coupled full-scale calculation.

Managing this calculation scheme as a whole is an important part of reactor analysis, and profound understanding of the methods, theory and underlying physics is absolutely essential for the safe and reliable utilization of nuclear energy. This project aims to enhance the knowledge basis needed for performing independent safety analyses for Finnish power reactors, relying on a novel approach using the continuous-energy Monte Carlo method for spatial homogenization. The work extends the methodologies applied in the calculation scheme to state-of-the-art and beyond. The development platform for the proposed research is the Serpent Monte Carlo code, which has been successfully developed for reactor physics applications at VTT since 2004.

1.1 Background and state-of-the-art

The proposed project continues the development of the Serpent Monte Carlo code, which was started at VTT in 2004. [1] Serpent is currently distributed free of charge for research and educational use by two data centers, the OECD/NEA Data Bank in Europe and RSICC in the U.S, and it has an international user community of some 600 users in 162 organizations in 37 countries around the world. The work has been funded from the previous SAFIR research programmes, in particular the KÄÄRME project in SAFIR2014, during which a new and improved code version, Serpent 2, was developed and made available to the user community. The successful work carried out within the previous SAFIR programmes is also recognized in the SAFIR2018 Framework Plan, where it is stated that the development should be continued and the range of application targets broadened.

Serpent development is currently divided into three main topics:

1. Advanced methods for spatial homogenization
2. Coupled multi-physics applications
3. Broadening the scope of applications beyond reactor physics

First of the three topics continues the work started in the KÄÄRME project, and it also forms the basis of the proposed research. The MONSOON project essentially combines the use of the continuous-energy Monte Carlo
method for neutron transport calculations into spatial homogenization and group constant generation for
deterministic full-core calculations.

Continuous-energy Monte Carlo simulation can be considered the golden standard for particle transport
applications. The calculation is based on the best available knowledge on neutron interaction physics without
major approximations, and the geometry can be modeled in three dimensions and refined to an arbitrary level of
spatial detail. In nuclear engineering, Monte Carlo codes have been traditionally used for applications like
criticality safety analyses, radiation shielding, detector modeling and validation of deterministic transport codes.

Spatial homogenization, however, requires calculation techniques that are not commonly available in general-
purpose Monte Carlo particle transport codes, and group constant generation for full-scale core simulations is
traditionally based on deterministic transport methods, such as collision probability or the method of
characteristics. Homogenization is typically performed at the fuel assembly level in two dimensions, which can be
considered a valid approach for traditional LWR applications with conventional fuel types, when the heterogeneity
of the core is limited to radial dimensions. This approach, however, is compromised in regions located near
control rod tips and when dealing with modern fuel types with axial profiling and partial-length fuel pins. The
methodology has been shown to break down completely in high-conversion LWR calculations, in which the core
consists of multiple separate seed and blanket regions laid on top of each other. [2,3]

Moving from deterministic transport codes to Monte Carlo simulation has several advantages for spatial
homogenization. The capability to use continuous-energy cross section data removes the need for intermediate
steps performed by deterministic codes to account for self-shielding effects, which considerably simplifies the
calculation scheme as a whole. Consequently, the same code and cross section libraries can be used for
modeling any fuel or reactor type without any application-specific limitations. The fact that the transport simulation
is inherently three-dimensional makes it possible to account for axial heterogeneities in homogenization, which
enables more rigorous calculations at the core level. The same Monte Carlo code used for group constant
generation can also be used for obtaining a full-scale reference solution without additional sources of error or
uncertainty, which can be considered a major advantage for the validation of the calculation sequence.

Serpent is one of the first Monte Carlo codes designed from the beginning for the purpose of spatial
homogenization. The transport simulation is optimized for performance in lattice calculations, and the code has
built-in calculation routines capable of producing all group constants needed for full-core nodal diffusion
calculations. The basic methodology was completed and successfully put to practice during the first two years of
the MONSOON project. [4-5] The results have demonstrated that the Monte Carlo based calculation sequence
can be used for producing the full set of group constants for PWR fuel cycle simulations at an acceptable
computational cost. [6-7] Similar calculations have been carried out in several Serpent user organizations, with
equally encouraging results (the complete list of publications is available at the Serpent website [1], for examples
of recent work see Refs. [8-12]).

The original research plan covered several topics and tasks related to the comprehensive validation of the
Monte Carlo based calculation sequence, as well as innovative methods for improving the results of nodal
calculations. Unfortunately, many of these tasks had to be postponed to later years as the project volume was
considerably reduced in 2015 and 2016. The importance of the work was nevertheless recognized within the
reference group. Additional funding was obtained from one of the member organizations (Fennovoima) for
completing work related to VVER-reactor calculations that had to be dropped from the 2016 project plan due to
budget cuts.
The project plan for MONSOON 2017 continues the work related to VVER reactors, in close in-kind collaboration with the Helmholtz-Zentrum Dresden-Rossendorf (HZDR). Validation studies are also extended from PWR to BWR applications, which essentially means covering a much larger state-point matrix in group constant generation. This work is carried out in in-kind collaboration with Westinghouse Sweden. The applications are also extended from steady-state fuel cycle simulations to transients, in which the computational challenges are slightly different. This part is done in close collaboration with the reactor dynamics project SADE.

The overall goal of using Monte Carlo codes for spatial homogenization is not just to replace deterministic lattice transport codes in the multi-stage calculation sequence, but to also exploit the inherent advantages of the method. The work planned for 2017 therefore includes development of new nodal diffusion solvers for testing advanced 3D methodologies. Some preliminary work and screening studies have already been carried out in 2016 in collaboration with the SADE project. [13] The topic has been discussed and is strongly supported by the reference group. The work is included in a new sub-task in WP1, and the project volume is correspondingly increased. Work on temperature-coupled burnup calculations and their impact on spatial homogenization is also continued from 2016 to 2017. Collaboration with Serpent user organizations has proven an invaluable resource for the development work, and similar to previous years, one work package is reserved for international collaboration.

1.2 Objectives and expected results

The project continues the development of the Serpent Monte Carlo code, started in 2004, and carried out within the previous SAFIR programmes. Compared to the KÄÄRME project in SAFIR 2014 the work in MONSOON is clearly more focused on spatial homogenization, and 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 improved methodologies are thoroughly validated and put to practice in the calculation schemes used at VTT for the safety analyses of Finnish power reactors.

The specific objectives and expected results from the beginning of the project can be summarized as follows:

1. Completing the methodology used in Serpent 2 for spatial homogenization (completed in 2015)
2. Implementation of an automated burnup sequence capable of covering the full state-point matrix in a systematic and automated manner (completed in 2015-2016)
3. Practical demonstration of the implemented capabilities for PWR fuel cycle simulations (completed in 2015-2016)
4. Practical demonstration of the implemented capabilities for VVER fuel cycle simulations (postponed due to budget cuts, started with the help of additional funding in late 2016)
5. Practical demonstration of the implemented capabilities for BWR fuel cycle simulations (postponed due to budget cuts, included in the 2017 project plan)
6. Practical demonstration of the implemented capabilities for PWR and/or BWR transient simulations (postponed due to budget cuts, included in the 2017 project plan)
7. Development of methodology to account for fuel temperature feedback in group constant generation (ongoing, included in the 2017 project plan)

8. Development of a new nodal diffusion solver to extend the current nodal modelling capabilities and to fully exploit the advantages of Monte Carlo based 3D homogenization (new topic, included in the 2017 project plan)

In addition to these specific objectives the project plan includes tasks involving source code maintenance and international collaboration. Many of the planned validation tasks cannot be accomplished without the efforts of active Serpent user organizations.

1.3 Exploitation of the results

Developing Serpent from a viable into a practical tool for spatial homogenization enables the code to be used as part of the reactor physics calculation scheme in a routinely manner. For Finnish end users this means replacing deterministic lattice codes, often used as a black box, with a domestic Monte Carlo code developed at VTT by a team with source-code level understanding of the underlying physics and methodology. Providing a versatile and easy-to-use calculation tool for group constant generation also considerably simplifies educating new experts in reactor core simulations.

The new methodologies for 3D homogenization and nodal diffusion calculations lead to more reliable computational analyses. The fact that the Monte Carlo method can also be used for providing high-fidelity reference solutions for full-core calculations means that the accuracy of the entire calculation sequence can be assessed in a whole new way, without additional uncertainties arising from nuclear data or methodological factors. Before such analysis was possible, the accuracy of some of the existing methods was considered sufficient, as it was deduced on the best available data at the time that other sources of error dominated in the calculation sequence. However, analyses utilizing Serpent 2 have shown that there is considerable potential for refining several existing methods.

The results of the MONSOON project are already exploited for research purposes in several Serpent user organizations worldwide. Use for licensing and industrial applications can be foreseen in the future, although this requires considerable effort for code validation in close collaboration with the user community.

1.4 Appropriateness of the project to SAFIR2018 programme

The Serpent code, together with the fuel cycle simulator and transient analysis codes developed at VTT and STUK, form a complete and independent calculation sequence for the safety analyses of Finnish power reactors. The continuous-energy Monte Carlo method provides a novel approach to spatial homogenization, enabling more rigorous modeling of conventional, as well as advanced fuel types and axially heterogeneous cores.

The work supports other on-going projects in SAFIR 2018, in which Serpent is involved as a calculation tool for various applications, in particular:

- SADE (VTT) Serpent is used for group constant generation for TRAB3D and HEXTRAN transient reactor analysis codes and as a reference for analyzing the accuracy of the codes and the results of improvements implemented into the codes.
• PANCHO (VTT) – The FINIX fuel behavior code developed in the project is internally coupled to Serpent.

• KATVE (VTT) – Serpent is used for criticality safety analyses and calculating radioactive inventory and decay heat source terms. The project also involves development of photon transport mode in Serpent for the purpose of radiation shielding applications in spent fuel storage facilities.

The international success of the Serpent code is recognized in the SAFIR2018 Framework Plan (Sec. 3.3.4.4.), where it is also explicitly stated that the development should be continued and the range of application targets broadened.

1.5 Education of experts

Currently the typical Serpent user is a university student, applying the code as a part of academic research and thesis work. Almost 400 scientific journal and conference papers and a total of 60 Ph.D., M.Sc. and B.Sc. theses and other student projects have been completed worldwide on Serpent-related topics since 2007. In Finland, the work has produced five doctoral and several Master's degrees, with more to be completed within the near future. The MONSOON project covers one of the three main topics for the future development of Serpent, and contributes to the education of new experts by providing research topics for students in Finland and abroad.

One member of the four-member Serpent developer team (Viitanen) received his doctoral degree during the first year of the project, and two members (Valtavirta, Kaltiainenaho) are working on their doctoral theses on Serpent-related topics. An M.Sc. thesis related to the Serpent-TRAB3D code sequence was completed in 2016, [13] and co-funded between the MONSOON and SADE projects. All key personnel apart from the project manager are young professionals under the age of 35. Two project group members (Sahlberg and Rintala) are starting their doctoral studies on the development of advanced nodal diffusion methods. The project manager holds the title of Adjunct Professor at Aalto University.

The educational impact of the project is not limited to students. Covering the complete reactor physics calculation scheme improves the knowledge basis necessary for performing reliable safety analyses for Finnish power reactors. Educating new experts for VTT compensates for the losses caused by retirement and brain drain to utilities and the regulator during the past ten years.

2. Work plan

The project has proceeded in stages, starting with the continuation of the work carried out in the KÄÄRME project of SAFIR2014. The first goal, accomplished for the main part in 2015-2016, was to develop Serpent into a practical tool for group constant generation, in other words, such that Serpent can replace current deterministic lattice transport codes in the traditional reactor physics calculation scheme. The work in WP1 and WP2 has
mainly been focused on the development and validation of computational framework used for group constant generation for reactor simulator and transient codes currently used at VTT:

1. **ARES** – Steady-state nodal diffusion code for the fuel cycle simulations of square-lattice LWR's, developed at STUK
2. **TRAB3D** – Time-dependent nodal diffusion code for the transient analyses of square lattice LWR's, developed at VTT
3. **HEXTRAN** – Time-dependent nodal diffusion code for the transient analyses of hexagonal lattice reactors, developed at VTT
4. **HEXBU** – Steady-state nodal diffusion code for the fuel cycle simulations of hexagonal lattice reactors, developed at VTT

One of the major accomplishments of the MONSOON project in 2016 was the practical demonstration that Serpent can be used for producing the full set of group constants for PWR fuel cycle simulations carried out using the ARES code at an acceptable computational cost. [7] This first-of-a-kind demonstration can be considered a significant milestone towards Monte Carlo based homogenization. The remaining two years of the project are focused on other nodal diffusion codes, reactor types and applications.

Work on the TRAB3D and HEXTRAN codes has already been started in collaboration with the SADE project [13-16]. So far the applications have been limited to steady-state calculations, and in 2017 the work is continued to transient analysis. Dynamic calculations using Serpent-generated group constants have been carried out using the Apros code [17], albeit using a simplified model without thermal hydraulics feedback. In addition to in-house nodal diffusion codes, increasing effort is devoted to calculation sequences involving nodal codes PARCS, POLCA8 and DYN3D with large international user basis. For the main part this work is carried as in-kind collaboration, by supporting the development and validation efforts at Serpent user organizations, in particular Westinghouse Sweden and HZDR.

The various comparisons between full-core nodal diffusion calculations and reference Serpent 3D solutions have revealed certain weak points and limitations in the methods used in VTT's nodal codes. This, together with the publication of insofar proprietary calculation methods [18] has raised serious discussion on the need to renew the methodology used for fuel cycle and transient calculations altogether. With the support of the Reference Group it was decided to study the potential of state-of-the-art methods by starting the development of a new neutronics solver based on 3D nodal methods. This new task in WP1 also serves the purpose of educating new experts in reactor safety analyses, as the work is carried out by two doctoral students.

Other topics, carried out in parallel with group constant generation, include accounting for the effects of fuel temperature feedback on assembly burnup calculations, and novel methods for cross section parametrization. The work began in 2015-2016 with a study involving coupled Serpent-ENIGMA burnup calculations, providing realistic assembly- and pin-wise temperature profiles over the irradiation cycle. This calculation system was applied to the burnup calculation part of the group constant generation to estimate the effects that the typical simplifications regarding fuel temperature distributions (flat radial profile, uniform temperature for all pins and through history) have on the generated group constants and end-of-life nuclide inventories.

Interaction with Serpent users has been found extremely valuable for the purpose of code development, since many of the user organizations share the same interests (in-kind collaboration with Westinghouse and HZDR was
The tasks include maintenance of source code and on-line resources, preparation of new cross section libraries for Serpent 2, organization of the 2017 Serpent User Group Meeting, daily interaction with the users and international collaboration. The importance of source code maintenance and other supporting actions is emphasized in 2017, since another major project, the Academy of Finland Nuclear Multi-Physics (NUMPS), ended in August 2016, and Serpent development currently relies almost exclusively on funding from SAFIR2018.

### 2.1 Work package 1 (WP1) – Development

This work package covers development of calculation methods for spatial homogenization, Serpent burnup calculations with fuel behavior code coupling and advanced nodal diffusion methods. The first two years of the project mainly focused on conventional methodology and in 2017 the work proceeds to new topics, including 3D methods for homogenization.

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<th>Partners in WP1</th>
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#### 2.1.1 Task 1 (T1.1) – Methods for spatial homogenization

The methodology used in Serpent 2 for traditional 2D homogenization was completed during the first year of the project. The work continued the development started in the KÄÄRME project in SAFIR2014. Serpent now has the capability to produce all input parameters needed for fuel cycle simulation and transient calculations performed with the nodal diffusion codes used at VTT. In 2017 the work continues more in the direction of 3D methods and capabilities needed for work related to the development of the new nodal diffusion solver in T1.4.

#### 2.1.2 Task 2 (T1.2) – Automated burnup sequence

The automated burnup sequence capable of covering all required state points with built-in branch capability was for the most part completed during the first two years of the project. The methodology was put to practice in PWR fuel cycle simulations. In 2017 the capabilities are extended to BWR's and transient simulations. In practice this means expanding the state point matrix to cover the operating conditions encountered in these applications.

Group constant output from Serpent runs is converted into library files read by fuel cycle simulator and transient analysis codes using a separate processing script. The development of such script, named SXSFit, was started in 2014. The script has now full support for the ARES fuel cycle simulator code, and some preliminary work has been done to support also TRAB3D, HEXTRAN, HEXBU and PARCS codes. The work continues in 2017, and new capabilities are added when they are needed for completing other tasks within this project.

This task also includes in-kind collaboration with Westinghouse Sweden and HZDR by supporting the validation efforts carried out for their calculation systems. In practice this means ensuring that all required input parameters for the nodal codes are readily available from automated Serpent runs.
2.1.3 Task 3 (T1.3) – Assembly burnup calculation with fuel temperature feedback

This task evaluates the effects of the simplifications made regarding the assembly fuel temperatures in the generation of group constants for steady-state and transient calculations.

The coupled burnup capabilities implemented in this task in 2015-2016 make it possible to run the history (burnup) part of the group constant generation simulation using either realistic fuel temperature profiles or the simplified constant and uniform fuel temperature distribution typically used in group constant generation. The effects of this simplification on nuclide inventories and group constants (and even on the final results of the simulator calculations) can then be explicitly estimated.

As the group constant generation for fuel cycle simulations moves to BWR’s (tasks 1.2 and 2.1) the effects of the simplification may be different than in PWR’s (which were covered in 2016). For example, the control rod history calculation deals with an assembly that is being burned with the control rod inserted into the unit cell. As the control rods in BWRs occupy one corner of the fuel assembly, they introduce an asymmetric perturbation to the neutron flux distribution in the assembly, and consequently to the pin power distribution. It is evident that the use of a homogeneous fuel temperature throughout the assembly (as is typically done) will overestimate the fuel temperatures in the corner closest to the control rod and underestimate the temperatures at the opposite corner. In 2017 the coupled burnup methodology will be applied to BWR group constant generation.

The coupled burnup scheme of Serpent currently uses the British fuel performance code ENIGMA as the fuel behavior solver. The use of the FINIX fuel behavior module developed at VTT would have been preferred as:

1. FINIX has been directly developed for multi-physics and coupled code purposes.
2. Modifying the FINIX code base for future applications will be much more straightforward (code developed in-house, ongoing dynamic development).
3. The use of FINIX would further leverage the national reactor analysis code base.

However, in the beginning of this task, FINIX was unable to model slow time scale changes in the nuclear fuel and the cladding due to fuel burnup and irradiation of the materials. For this reason, ENIGMA has been used in the coupled burnup methodology thus far. The capabilities of FINIX are being extended in the PANCHO project to model the effects of irradiation of the fuel and cladding. This means that the only reason for choosing ENIGMA over FINIX will be removed. For this reason, ENIGMA will be replaced with FINIX in the coupled depletion scheme with Serpent during 2017. This will require minor changes to the current coupling with FINIX, mainly related to supplying FINIX with the detailed burn up and fast neutron flux distributions.

Performing coupled burnup calculations involves additional challenges related to discretization errors and the stability of the iteration scheme. Recent progress in the development of burnup algorithms based on sub-step methods have shown promising results [19,20] and the methodology will be implemented in Serpent as well.

2.1.4 Task 4 (T1.4) - Development of an advanced nodal diffusion solver

The use of Serpent 2 both for generating group constants and providing high-fidelity reference solutions has enabled rigorous analyses of the existing nodal diffusion methods used in VTT’s calculation codes. These analyses have revealed inaccuracies, which can be considered significant compared to the other sources of error in the calculation sequence [14-16].
Before such analysis was possible, the accuracy of the current nodal diffusion solvers was considered sufficient, as it was deduced based on the best available data at the time that the other sources of error dominated the calculation chain. In the light of the new information, the development of a more refined nodal solver is topical. This is further highlighted by the fact that several of the other significant sources of inaccuracies are being reduced with the development of the FINIX fuel behavior module in the PANCHO project and the coupling of porous flow CFD code PORFLO to nodal diffusion codes in the SADE project. Consequently, fuel modelling and thermohydraulics can no longer be assumed to dominate the computational error.

There is significant evidence in existing literature that the inaccuracies found in the current nodal solvers can be eliminated almost completely with a more advanced nodal solver [21,22]. This is also supported by the results of the M.Sc. thesis produced jointly by MONSOON and SADE in 2016 [13], and by the more recent work in the SADE project involving the modelling of control rod tips. A more advanced nodal solver would also enable the utilization of the inherent advantages of Serpent 2 as a three-dimensional Monte Carlo homogenization tool, which could offer unique advancements in research and reactor safety analysis.

The benefits of developing a new nodal solver are not limited to technical details either. The currently used independent calculation tools were developed from the 1980's to the beginning of the millennium. The original developers of the solvers and their computational models have retired, moved on to other tasks, or are otherwise preoccupied by other assignments. This hinders the long-term maintenance of the solvers and poses significant limitations when it comes to implementing new methods and refinements. It is no longer practically feasible to modify the existing calculation tools in any significant manner. Maintaining source-code level understanding of the calculation methods is nevertheless considered vital for preserving the expertise required for performing reliable safety analyses for nuclear reactors.

Therefore it is proposed that the development of an entirely new nodal diffusion solver is started within the framework of this project. The aim of the task is to educate new experts in reactor core calculations and to lay the foundations for the long-term development of new solution methods for steady-state simulations and transient reactor safety analysis. However, the first phase of this development will be focused on demonstrating the benefits of a more sophisticated nodal calculation model for the new solver code, in order to justify further work on the topic.

In practice the task consists of choosing and implementing a new single node solution scheme in a three-dimensional square-lattice geometry. After the implementation is verified, the work continues to combining individual single-node solutions to a full core steady state calculation model. Once the full-scale model is ready for validation, the improved accuracy is demonstrated by comparison to existing nodal methods and Serpent 3D reference results. In this first stage the work does not yet include any optimization of computational efficiency, time dependence, state-point coupling or expanding the model to hexagonal lattice geometry, as the demonstration of accuracy is deemed to be the first priority.

2.2 Work package 2 (WP2) – Validation

Developing methods for full-scale fuel cycle and transient analyses based on the two-stage calculation scheme is a formidable task, which requires comprehensive validation of each calculation sequence. The general approach is to compare the results of the nodal code with Serpent-generated group constants to a full-scale 3D Serpent calculation, which can be considered the ideal reference case for validating the methodology.
The codes and challenges are slightly different for steady-state simulations and transients, and the work package is correspondingly divided into two tasks. In 2015-2016 the work was focused on PWR fuel cycle simulations and steady-state VVER calculations, and in 2017 the validation is continued to BWR’s and transients. The validation of calculation sequences involving VTT’s transient codes is carried out in collaboration with the SADE project.

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2.2.1 Task 1 (T2.1) – Group constant generation for fuel cycle simulations

Work on the Serpent-ARES code sequence started in the KÄÄRME project of SAFIR 2014 was continued in 2015-2016, using the MIT BEAVRS benchmark [23] as the test case. The calculations included initial core HZP and HFP calculations and simulation of operating cycle 1. [5-7] The results were compared to experimental measurements provided with the benchmark specification. The calculations could be continued to cycle 2, but at the time of this writing this is not considered a high priority compared to the other, more urgent validation tasks. Validation of Serpent-HEXBU code sequence had to be dropped from the 2016 project plan due to budget cuts. However, additional funding from Fennovoima in late 2016 allowed performing preliminary calculations for VVER-1000 HZP initial core configuration.

A new validation task for 2017 involves in-kind collaboration with HZDR, Germany, in which an EU project “Institutional and technical cooperation with Gosatomnadzor to develop its capabilities on the basis of transferred European safety principles and practices” is currently under way. This project includes validation of the Serpent-DYN3D calculation scheme using the X2 AER benchmark as the test case. The benchmark data contains detailed description of Khmelnitsky-2 VVER-1000 unit initial core and first 4 fuel cycles, including results of start-up experiments, power distributions at different states and boron let-down curves. Also included are measurements made during xenon oscillation, main pump trip and fast power decrease transients, which can be used validating transient simulations. VTT’s role in this benchmark is to support the work carried out at HZDR by ensuring that all parameters required by DYN3D are readily available.

Another new topic for 2017 is the use of Serpent for generating group constants for BWR calculations. This task involves in-kind collaboration with Westinghouse Sweden, who have expressed considerable interest in adopting Serpent as a computational tool for BWR analyses, in particular for producing group constants for the POLCA8 nodal diffusion code. The collaboration includes sharing experience and expertise, which for VTT means hosting an M.Sc. student from Westinghouse for a period of several months.

2.2.2 Task 2 (T2.2) – Group constant generation for transient simulations

The work on Serpent-TRAB3D code sequence started in the KOURA project of SAFIR 2014 [14] and continued in the SADE and MONSOON projects of SAFIR 2018. [15] A related M. Sc. thesis (Sahlberg) [13] was completed in 2016. However, because of the budget cuts in 2015-2016, the work has not yet been extended to actual transient simulations.
In 2017, a suitable test case is selected from the reactor dynamics validation matrix, and the calculations are repeated with Serpent-generated group constants. This may involve a VVER-, PWR- or BWR-core, and the transient calculations are correspondingly carried out using TRAB3D or HEXTRAN. The results are compared to reference data in the benchmark specification, and if available, results from previous studies.

It is possible that the validation of the Serpent-DYN3D calculation sequence at HZDR described in Sec. 2.2.1 is extended to transients within the framework of the X2 AER benchmark.

2.3 Work package 3 (WP3) – Serpent user community

This work package covers international collaboration and daily interaction with the Serpent users. Forms of interaction include maintaining the Serpent website and discussion forum, organizing annual user group meetings and participation in the activities of the international reactor physics community. Travel costs are allocated for participation of Serpent developers and project group members in the 7th International Serpent User Group Meeting, other invited workshops, as well participation in OECD/NEA working groups and ANS RPD meetings.

Serpent 1 is currently distributed free of charge for non-commercial research and educational use by the OECD/NEA Data Bank and RSICC. Serpent 2 has been developed since 2010, and currently distributed by request to registered users of Serpent 1. Most of the users have already adopted the new version, and the code is practically ready for official public distribution. The procedures needed for preparing Serpent 2 for distribution are included in this work package, and the original plan was to submit the code to the OECD/NEA Data Bank and RSICC by the end of 2015. The public release of Serpent 2 has been postponed several times due to the reductions in the project volume. The on-going tasks, namely the preparation of an input manual in the form of an on-line Wiki and the generation of new ACE format cross section libraries are still considered vital for the future of the Serpent code, and therefore kept in the project plan for 2017 as well.

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2.3.1 Task 1 (T3.1) – Serpent Wiki and daily interaction with users

A Wiki-based on-line platform [23] was set up for Serpent in late 2015. The Serpent Wiki serves as an input manual as well as a repository for example inputs and validation data. The Wiki is maintained and expanded in 2017. This task also covers the maintenance of Serpent website and support provided to Serpent users via e-mail and the Serpent discussion forum.

2.3.2 Task 2 (T3.2) – Generation of new cross section libraries for Serpent 2

New ACE format cross section libraries have been prepared for Serpent 2 from the most recent evaluated nuclear data files, such as ENDF/B-VII.1, JENDL-4.0 and JEFF-3.2, using the NJOY nuclear data processing system. The data includes neutron interaction cross sections, as well as photon cross sections for the recently developed gamma transport mode (this work connects to the KATVE project in SAFIR 2018). The new cross section libraries
are currently being extensively tested by comparison to MCNP6, which shares the same ACE library format. The work continues in 2017.

2.3.3 Task 3 (T3.3) – International collaboration and 7th International Serpent UGM

This task covers the organization of the 7th International Serpent User Group Meeting together with the host organization. The previous meetings were held in Dresden, Germany (2011); Madrid, Spain (2012); Berkeley, USA (2013); Cambridge, UK (2014); Knoxville, USA (2015) and Milan, Italy (2016). The next meeting will take place in the U.S (possibly hosted by University of Florida), but the date has not yet been fixed. This task also covers participation in the activities of international scientific societies and organizations, in particular the ANS Reactor Physics Division (Leppänen) and OECD/NEA working groups (Leppänen, Viitanen), as well invited workshops and seminars.

2.4 Work package 4 (WP4) – Project management

This work package covers project management and participation in reference group meetings.

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<thead>
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3. Deliverables and milestones 2017

Since the Serpent code has an extensive user basis, the results of this study are primarily published as scientific journal and conference papers, which are not only readily available to the user community, but also peer-reviewed, and as such valuable documents for validation purposes. Similar practice has been applied in other SAFIR2018 projects involving Serpent development. Some minor tasks are documented in the Serpent Wiki [23], which acts as an on-line user manual.

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### 4. Project organization

Project manager is Jaakko Leppänen and deputy project manager Ville Valtavirta. The entire project staff is employed by VTT.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organization</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
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<tr>
<td>Jaakko Leppänen, D. Sc</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>1.1, 1.2, 2.1, 3.1, 3.2, 4.1, 4.2, 4.3, 5</td>
<td>1.7</td>
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<tr>
<td>Ville Valtavirta, M. Sc.</td>
<td>Research scientist</td>
<td>VTT</td>
<td>1.3, 3.1, 4.2</td>
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<tr>
<td>Toni Kaltiaisenaho, M. Sc.</td>
<td>Research scientist</td>
<td>VTT</td>
<td>3.1, 4.2, 4.3</td>
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<tr>
<td>Ville Sahlberg, M. Sc.</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>1.4, 2.2</td>
<td>3.5</td>
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5. Risk management

The project has a good track record of completing all planned tasks. Even so, since the work is mainly related to the development of new and complex calculation methods, there are certain risks and uncertain factors as listed below.

1. Accomplishing many of the planned tasks does not depend only on the funding allocated to the MONSOON project, but also other projects in close collaboration, such as SADE, PANCHO and KATVE. Reducing the volume of these projects may require changes in the project plan.

2. Some of the tasks involving the FINIX fuel behavior solver depend on development work carried out outside this project. In case the required capabilities are not available in time, the studies in task 1.3 are continued with the Serpent-ENIGMA coupling.

3. Even though it is almost certain that adopting state-of-the-art methodologies in the new nodal diffusion solver will lead more accurate results compared to current calculation codes, there is a chance that this undertaking will fail or turn out to be unrealistic with the given resources. In such case the development of the code is discontinued, and focus on future work is turned to existing codes that are still actively maintained and developed, such as DYN3D and PARCS.

4. Some of the validation tasks involve in-kind collaboration with Serpent user organizations, which means that the possibilities to affect in the outcome are very limited. On the other hand, VTT’s role in these tasks is merely supportive, so there is no major financial risk.

5. The agreement with University of Florida to host the next Serpent UGM is preliminary only, and finding an appropriate time frame with universities following a fixed curriculum is always a challenge. In case the arrangement cannot be accomplished, the next UGM will be organized at Georgia Tech. or in Finland (the tentative location for 2018).
References


16. Sahlberg, V. “Recalculating the steady state conditions of the V-1000 zero-power critical facility at Kurchatov Institute using Monte Carlo and nodal diffusion.” In. 26th Symposium of AER on VVER Reactor Physics and Reactor Safety, 10-14 October, Helsinki, Finland.


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

<table>
<thead>
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<td>T3.2 Generation of new cross section libraries for Serpent 2</td>
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<td>T3.3 International collaboration and 7th International Serpent UGM</td>
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**TOTAL** 11.5 122 0 15 0 0 9 146 102 44 0

Comments:
Other costs include the use of VTT’s computer clusters at the rate of 5€ per budgeted working hour
SAFEIR 2018 Project plan

NURESA 2017
Development and Validation of CFD Methods for Nuclear Reactor Safety Assessment

Virtauslaskennan menetelmien kehittäminen ja kelpoistaminen ydinreaktorien turvallisuusanalyyyseihin

Timo Päätikangas, Juho Peltola
VTT Technical Research Centre of Finland

Vesa Tanskanen
Lappeenranta University of Technology

Tommi Rämä
Fortum Power and Heat Oy
Version control:

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<th>Version</th>
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<td>Timo Päätikkangas, Juho Peltola, Antti Timperi, Timo Siikonen, Vesa Tanskanen, Karoliina Ekström</td>
<td>Plan updated after the funding reductions for the SAFIR2018 Programme in 2015</td>
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<td>3.51</td>
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<td>Timo Päätikkangas, Juho Peltola, Timo Siikonen, Vesa Tanskanen, Karoliina Ekström</td>
<td>Plan amended according to the comments of the SG2 meeting 1/2016 on page 17</td>
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<td>4.2</td>
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<td>Timo Päätikkangas, Juho Peltola, Vesa Tanskanen, Tommi Rämä</td>
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<td>4.21</td>
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1. Research theme and motivation

1.1 Background and state-of-the-art

International evaluation of the SAFIR2010 programme was performed in 2010 [1]. On thermal hydraulics, the evaluation panel issued the following recommendation: “The growing role of CFD codes in the safety analysis and licensing and the unavailability of commercial code sources recommend considering the CFD code development as a Finnish or cooperative activity in the future.” In Finland, the main tool for Nuclear Reactor Safety (NRS) analysis was at that time the commercial ANSYS Fluent code. Fortum had started one year earlier a SAFIR-project on the validation and development of the open source Computational Fluid Dynamics (CFD) code OpenFOAM in cooperation with VTT, Aalto University and LUT.

International evaluation of the Finnish Nuclear Safety Research Programme SAFIR2014 was performed in the beginning of 2014 by an international panel [2]. On thermal hydraulics, the panel recommended, for instance, that CFD methods for solving two-phase flow problems should be validated against experiments.

In the Nordic Thermal Hydraulic Network (NORTHNET), three Roadmaps have been written on the development of numerical thermal hydraulic methods and performing experiments. In the NORTHNET Roadmap 1 [3], OpenFOAM is developed and experiments are performed for the modelling of dry-out in BWRs. In Roadmap 2, the coupling of thermal hydraulic codes with neutronics codes is improved. In addition, the low frequency oscillations in BWRs have been studied. The NORTHNET Roadmap 3 was revised recently [4]. The new Roadmap 3 concentrates on experiments and modelling of the stratification of the pressure suppression pools of BWRs. In addition, experiments and modelling of drywell and wetwell sprays is proposed. In particular, the effect of the wetwell sprays on the stratification of the pressure suppression pool is of interest.

In the SAFIR2018 Framework Plan [5], the research needs in different areas are described. Some of the guidance given in the Framework Plan on the Reactor Safety research area is summarized in the following. In the Framework Plan, it is stated that the risk related to the licenses of the analysis codes should be as small as possible. This is the case, when open source codes are used or the analysis code is owned by organization participating in the SAFIR-programme. The Framework Plan points out that in future it is also necessary to combine results of different computer codes or couple them directly with each other.

The importance of the validation of the codes is emphasized in the Framework Plan. No comprehensive validation matrix for CFD codes exists, but some recommendations on the validation have been given by OECD working groups. The procedures for the validation of CFD tools have to be developed so that CFD analysis can be used in licensing calculations of NPPs. Therefore, CFD grade validation data is needed that can be obtained, for instance, with Particle Image Velocimetry, tomography or wire-mesh sensors. The nationally important experimental data should be listed and the missing experiments should be performed in national or international projects. The validation of the numerical tools for analyzing Fluid-Structure Interaction (FSI) should be performed.

According to the Framework Plan, the development and validation of CFD tools for two-phase flows including boiling and condensation is a demanding long term goal. Since the resources in the national research programme are limited, international co-operation makes possible to achieve better results. Use, development and validation of the open source CFD code OpenFOAM should be continued in international co-operation.

The SAFIR2014 Reference Group 4 recommended in May 2014 [6] that a strategy for the use, development and validation of the open source CFD code OpenFOAM should be formulated. An Ad Hoc Group meeting on the OpenFOAM code was arranged [7], where the members of the Reference Group 4 and representatives of research organizations were invited. The Nuclear Reactor Safety (NRS) problems requiring CFD analysis with or without coupling to other codes were evaluated in the meeting. The relevant problems were taken from NEA reports assessing use of CFD for NRS [8,9]. A few items were added in the problem list by the Ad Hoc Group and the priority of the problems was determined. The main results of problem identification and ranking are summarized in the following for NRS problems requiring single phase CFD analysis and two-phase CFD analysis [10].

In Figure 1, the most important single-phase NRS problems identified by the Ad Hoc Group have been put on timeline. Mixing, stratification, boron dilution and hydrogen distribution are single-phase mixing problems. Thermal
fatigue, tube vibration and coupling with third party codes are related to fluid-structure interactions. Coupling with third party codes also involves coupling of CFD and system codes or CFD and neutronics codes.

In Figure 2, the most important two-phase NRS problems identified by the Ad Hoc Group have been put on timeline. Three of the items are related to boiling: Departure from Nucleate Boiling (DNB), dry-out and transition boiling. Two items deal with condensation: direct-contact condensation and waterhammer condensation. Reflooding consists of complicated three-dimensional flow pattern combined with complex physical phenomena. Pipe-break with in-vessel mechanical load deals with FSI together with two-phase phenomena: boiling and flashing.


The NURESA project proposal consists of four year Work Packages, where CFD methods are developed and validated for the identified most important topics in NRS assessment. In Work Package 1 (WP1), international single-phase mixing benchmarks are participated. In WP2, PPOOLEX spray experiments are modelled with CFD codes in co-operation with Swedish partners. In WP3, CFD models for DNB and dry-out are developed for OpenFOAM code in co-operation with international partners. In WP4, CFD-Apros simulations of NPPs are performed to validate the coupling of CFD and system codes. Coordination of the project is done in WP5.

1.2 Objectives and expected results

In WP1, CFD models are validated for single phase mixing and stratification calculations. This Work Package is directly related to the important "Mixing and stratification" item discussed in Section 1.1 (Figure 1). Similar numerical models are also used in the calculation of "Hydrogen distribution", “Pressure Thermal Shock” and “Boron dilution”. In WP1, single phase CFD models are validated for mixing and stratification calculations. In particular, the international benchmarks give as a result an understanding of the uncertainties involved in using CFD calculations for NRS assessment. As a result of WP1, improved understanding on the uncertainty of state-of-the-art CFD calculations on mixing and stratification is obtained.

In WP2, spray experiments proposed to be performed with the PPOOLEX facility at LUT are modelled in co-operation with KTH. Improved CFD models for condensation and evaporation of spray droplets are developed and validated. In addition, improved model for the film condensation on walls is developed. The interaction of the spray with the pressure suppression pool is studied. Accurate modelling of sprays is important, for instance, in modelling of “Hydrogen distribution” (see Figure 1). As a result of WP2, validated models for spray and for film condensation are obtained. In addition to calculations of containments, the developed models can also be applied to pressurizers of PWRs.

In WP3, open source CFD code OpenFOAM is developed and validated for NRS assessment. In particular, models for boiling and DNB in PWRs are developed. The work is done in co-operation with KTH, which develops models for dry-out in BWRs (see Figure 2). The long-term goal is, however, that OpenFOAM becomes publicly available, transparent and efficient CFD simulation tool that provides the capability to simulate a wide range of NRS applications. To achieve adequate user base for credible validation it is important that at least the most important two-phase models developed in WP3 are included in the official OpenFOAM release. This also makes possible the maintenance of the implemented models with a reasonable effort.

This goal of WP3 is achieved by co-operation with the OpenFOAM Foundation and international partners that have overlapping development targets. The national network of OpenFOAM users performing nuclear safety CFD analysis is further developed and the national toolset of application specific closure models, utilities and best practice guidelines is collected during the project.

In WP4, CFD-Apros coupling is validated, which is part of the topic “Coupling with third party codes” discussed in Section 1.1. The goal is to make possible more realistic simulations of large integral systems, where important three-dimensional components are accurately modelled. As a result validated methods for performing such simulations are obtained.

In WP5, co-ordination of the project is performed and international co-operation is done with KTH, HZDR, NORTHNET Roadmap 1 and Roadmap 3 and the OpenFOAM Foundation.

1.3 Exploitation of the results

The developed and validated CFD methods are used by the regulator, utilities and research organizations in NRS assessment. In addition to the validation of the CFD methods, the uncertainty of the results is assessed in international benchmark calculations. The information on uncertainties is essential in using CFD in NRS analysis.

The benchmark calculations performed in WP1 provide validation of the CFD methods in the calculation of mixing and stratification. In addition, information on the uncertainties of the results is obtained, when the results of several international partners are compared to experimental results.

The modelling of PPOOLEX experiments in WP2 provides improved understanding on the pressure suppression function of the BWR containment. New models for condensation and evaporation of spray droplets and liquid films can also be used in other NRS problems, such as in modelling of pressurizers of PWRs.

The subcooled boiling models of OpenFOAM that are developed and validated in WP3 can be used after the second year of the project for NRS assessment. A model for DNB will be available at the end of the four year project. A main result of WP3 is a publicly available, transparent and efficient open source CFD simulation tool for nuclear safety. In addition, a related national toolset is formed with the application of specific closure models and
best practices. This will provide a common software platform for national and international co-operation in the field of nuclear safety related CFD.

The validation of coupled CFD-Apros calculations in WP4 provides a new analysis tool for NRS assessment. The coupled calculations make possible to analyses large systems with three-dimensional components.

1.4 Appropriateness of the project to SAFIR2018 programme

The NURESA project proposal has been prepared based on the SAFIR2018 Framework Plan as is described in detail in Section 1.1. In addition, the expertise of the SAFIR2014 Reference Group 4 was used by arranging an Ad Hoc Group meeting, where the priority of the NRS problems requiring CFD analysis was determined. The Roadmaps shown in Figures 1 and 2 for the development and validation of CFD codes were outlined based on the analysis of the results of the Ad Hoc Group meeting. The work packages of the NURESA project proposal are designed to implement several of the important items listed in the Roadmaps.

1.5 Education of experts

Several young scientists work in the project and will be educated in nuclear reactor safety assessment. The research work included in the project will be a part of their doctoral thesis. M.Sc. Ville Hovi will perform modelling of sprays and condensation in PWR pressurizer. M.Sc. Giteshkumar Patel (LUT) will develop condensation models for the OpenFOAM solver. M.Sc. Juho Peltola (VTT) will develop models for Departure from Nucleate Boiling (DNB). M.Sc. Tommi Rämä (Fortum) will develop and test model for VVER-440 pressurizer. The number of experienced OpenFOAM users in Finnish organizations will increase.
2. Work plan of the NURESA project

In the following, the proposed work plan of the NURESA project is presented. The Work Packages of the project contribute to the implementation of the Roadmaps on the validation and development of CFD models that were presented in Section 1.1.

2.1 Work Package 1 (WP1) CFD benchmarks on mixing and stratification

Mixing of hot and cold water in the ROCOM test facility is calculated with coupled CFD-Apros calculation. A journal article is written on the simulation results.

ROCOM experiment PKLIIT1.1 has earlier been calculated in an internal development project at VTT by using Fluent-Apros coupling. The results obtained were not satisfactory because the CFD mesh and the nodalization in the Apros model were not fine enough. In addition, the chosen turbulence model did not perform well in the regions where the flow velocity was small and the level of turbulence was low. Therefore, re-calculation of the experiment is proposed in this Task.

No work is done in WP 1 during year 2017.

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2.1.1 Task1 (T1.1) ROCOM experiment PKLIIT1.1 on mixing in pressure vessel

An overview of the ROCOM test facility is presented in Figure 3. The test facility consists of a Reactor Pressure Vessel (RPV) in the middle, four loops connected to it, and holding tanks for water treatment. Since the RPV, shown in Figure 4, is made out of transparent acrylic glass, the tests are conducted at atmospheric pressure. To neglect heat losses, all the tests are performed at room temperature. Density differences are obtained with different mixtures of water, sugar and alcohol. Measurements are based on conductance (Wire Mesh Sensors, WMS), which means that concentration differences of salt tracer are measured instead of temperature differences.

Experiment PKLIIT1.1 was chosen because it is long enough to allow the pulse of “hot” water to pass through the RPV and the main circulation loops, i.e., through both calculation domains of the coupled simulation. It corresponds to the transient that follows the reconnection of an isolated main circulation loop in the aftermath of a main steam line break, in which a slug of warm water enters the RPV from a previously isolated steam generator. An overview of the test setup and the locations of the WMSs are presented in Figure 4.

During the initial steady-state, Loops 1, 2 and 4 are in normal operation, with a constant flow rate of 3.3 l/s, while Loop 3 is isolated. The whole primary circuit is at a uniform density (corresponding to room temperature 20 °C). At the beginning of the transient (at 0 s), relatively light water (corresponding to 79 °C) is injected into the cold leg of Loop 3 with a flow rate of 2.14 l/s. The excess water flows to an overflow tank, which is connected to the hot leg of Loop 1.

Coupled Fluent-Apros model constructed earlier will be used for the calculation, where the necessary refinements of the CFD mesh and Apros model will be made. The $k$-$\omega$ SST turbulence model will be used for the modelling of turbulence. Additional source terms for the turbulence caused by buoyancy will be included in the turbulence model. The source terms have been implemented in Fluent already earlier at VTT. Injection Loop 3, will be included in the CFD model; other loops will be described with Apros model. The coupling feature of Apros and Fluent developed earlier by VTT and Fortum will be used in the simulation.

A journal article will be written on the simulation results.
2.2 Work Package 2 (WP2) CFD modelling of PPOOLEX spray experiments

Experiments performed with the PPOOLEX experimental facility at LUT will be modelled with CFD simulations. PPOOLEX is a down-scaled model of BWR containment. The effect of the dry well and wet well spray on the mixing and stratification in the containment will be modelled. Work Package 2 is therefore related to the single phase mixing problems of the Roadmap presented in Section 1.1. The focus in this work package will be on the effect of spray on the mixing and stratification in the gas space of the containment and in the pressure suppression pool. The effect of spray is also important in the analysis of hydrogen distribution in the containment. Therefore, this Work Package is also related to the item “Hydrogen distribution” in the Roadmap of Figure 1.
2.2.1 Background

BWR containment is a complex system that includes such typical elements as pressure suppression pool, spray and containment venting systems for containment pressure control, blowdown pipes for rapid steam condensation in case of LOCA, spargers for the vessel pressure relief valves, strainers for water supply to emergency core cooling and spray systems, nozzles and strainers of the residual heat removal (RHR) system, vacuum breakers, etc. COPSAR-project (Containment Pressure Suppression Systems Analysis for Boiling Water Reactors) has been proposed to NKS, where LUT, VTT and KTH investigate the phenomena that can affect pressure suppression function due to the operation of the different systems in the BWR containment.

There are several scenarios of safety importance where containment pressure suppression function and pressure suppression pool operation are affected by (i) stratification and mixing phenomena, (ii) interactions with emergency core cooling system (ECCS), spray, residual heat removal (RHR), filtered containment venting system (FCVS), (iii) overall water balance in the containment compartments, and (iv) interplay between pool behavior, diagnostics and procedures.

Specifically the above scenarios include (i) different LOCAs including scenarios with steam line break inside the radiation shield, broken blowdown pipes, and leaking safety relief valves; (ii) station blackouts; (iii) severe accidents. There is a need for validated tools for simulation of realistic accident scenarios with interplay between phenomena, safety systems, operational procedures, and overall containment performance.

It has been suggested that mixing induced by spray had a role in the pressure drop in Fukushima Daiichi Unit 3 where pressure build-up in the containment during the first 20 hours after station blackout was attributed to stratification in the pool. Formation and disappearance mechanisms of thermal stratification are therefore under investigation [13]. Addressing stratification and mixing issues in a large pool is thus important and additional data on pool behavior are needed for the validation of computer models and realistic evaluation of safety margins.

The COPSAR NKS-project is planned to consist of the combined effort by LUT, VTT and KTH to implement the ideas outlined in the recently revised NORTHNET Roadmap 3 document [4]. The work at VTT and LUT is proposed to be done within the NURESA and INSTAB projects of the SAFIR2018 programme. The work at KTH will be done in the project “Modelling of Stratification and Mixing Transients in a BWR Pressure Suppression Pool” supported by NORTHNET Roadmap 3 and “Analytical support for the OECD/NEA HYMERES project” supported by SSM. The combined effort will be coordinated by LUT within the COPSAR NKS-project and guided by the NORTHNET Roadmap 3 Reference Group.

The proposed NKS-project COPSAR will deal with several phenomena and scenarios that are mentioned above. At LUT, the aim will be to experimentally study the interplay between pool behavior and spray systems, stratification due to a leaking safety relief valve or small LOCA and the effect of RHR system operation on pool mixing. Although several full scale experiments have been done on wetwell pool mixing due to pressure relief system blowing and activation of systems for forced mixing, limited data is available, for example, on the details of pool mixing due to activation of the wetwell spray systems. Additional data about pool interactions with spray systems are needed for the realistic evaluation of safety margins in designs and for validation of computer models.

Effective Heat Source (EHS) and Effective Momentum Source (EMS) models have been developed by KTH for simulation of steam injection into a pool. In this work, we will further extend the concepts of the EHS/EMS to spargers, strainers, RHR system nozzles, and operation of blowdown pipes with non-condensable gases [14]. Simulation tools will be developed for the modelling of spray operation, formation of liquid films on the vessel wall, stratification of the water pool and gas space. Predictive capabilities of the GOTHIC code in modelling of mixing and stratification will be further evaluated against experiments with spray.

VTT will participate in the COPSAR project by performing CFD calculations of the experiments performed at LUT with the PPOOLEX facility. The effects of the dry well and wet well sprays will be modelled with CFD calculations. In particular, the effect of the wet well spray on the thermally stratified water pool will be studied together with the mixing caused by the RHR system. The subtask of VTT in the COPSAR-project is proposed to be performed in Work Package 2 of the present NURESA project.

2.2.2 Targets and results

The COPSAR-NKS project aims to increase understanding of the phenomena related to BWR pressure suppression function to enhance capabilities to analyses Nordic BWR containments under transient and accident conditions. Particularly, additional information is needed on

- spray efficiency in mixing of stratified gas layers
- feedbacks between wetwell water pool and spray
- formation of liquid films on the vessel wall due to spray operation and their effect on heat transfer and local condensation and heat flux to the pool
- effect of increased spray water temperature on spray efficiency in case of losing spray water cooling
- effect of spargers, RHR nozzles, strainers, vacuum breakers and blowdown pipes on mixing and stratification of the pool.

To achieve the project objectives, a combined experimental/analytical/computational program is proposed. LUT will be 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. VTT and KTH will use the gathered experimental database for the development, improvement and validation of numerical simulation models. Also analytical support will be provided for the experimental part by pre- and post-calculations of the experiments.

In 2015, CFD calculations were performed for one of the spray nozzles tested at LUT for the installation in the PPOOLEX facility. Model for the size distribution of droplets was chosen based on the available experimental and literature data. The results of the CFD calculations were compared to the shadowgraphy data obtained at LUT. Pre-calculations of the PPOOLEX spray experiments were performed, where four spray nozzles were installed in the wet well of the PPOOLEX facility.

In 2016, CFD simulations of PPOOLEX experiments on SRV spargers have been performed. First, CFD simulations of the condensation of clouds of small vapor bubbles were performed. Then thermal stratification of the pool during steam injection was studied with CFD calculations by using source terms for mass, momentum and enthalpy. Stratification and mixing of the pool in the PPOOLEX experiment SPA-T1 was calculated. The results are being compared to the experiment by the end of year 2016.

Partners and person months allocated to WP2 in 2017 are given in the table.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>4.5</td>
</tr>
</tbody>
</table>

2.2.3 Task 1 (T2.1) Condensation of bubble clouds

In 2017, pre-calculations will be performed with ANSYS Fluent for the small-scale separate effect facility, where steam will be injected through a single orifice into water pool. First, CFD model for the small-scale facility will be constructed. The Euler-Euler method with two-resistance condensation model of Fluent will be used. The calculated condensation rate and penetration of the vapor jet into the pool will be calculated and later compared to the experimental data. The effective momentum and heat sources generated by the steam injection into the water pool will be determined. The results will be compared to the EMS/EHS model of KTH.

This Task will be done in co-operation with the CFD team of LUT.

2.2.4 Task 2 (T2.2) Calculation of stratification and mixing

In 2017, a spray experiment performed at PPOOLEX will be calculated with ANSYS Fluent. The water pool will be modelled with the Euler-Euler model of Fluent, where droplets will be described with the Discrete Particle Model (DPM). The effect of the spray droplets on the stratified pool will be calculated. The mass, momentum and enthalpy sources from the droplets to the pool water will be described with user-defined functions of Fluent. The possible deterioration of the thermal stratification of the pool will be studied. The results will be compared to PPOOLEX experiment. The connections to the stratification in Fukushima Daiichi Unit 3 will be discussed.
2.2.5 Year 2018

The spray experiments performed in the dry and wet well will be calculated. Detailed model for the liquid-water film on the walls of the containment will be constructed. The Eulerian film model of ANSYS Fluent will be used as the starting point, where the condensation and evaporation models will be added. The interaction of spray droplets with the film will be included in the model. The spray experiments performed with the PPOOLEX facility will be calculated. The interaction of the wet well spray with the thermally stratified pressure suppression pool will be investigated. The studies on the break-up of the stratification caused by cold spray water and the suction of water from the pool will be continued.

2.3 Work Package 3 (WP3) OpenFOAM solver for nuclear reactor safety assessment

In WP3, two-phase OpenFOAM solvers are developed and validated for boiling (DNB) and direct contact condensation. The work on boiling is done in co-operation with KTH, where OpenFOAM models of dry-out are developed and validated. Work package focusses on the important two-phase flow of NRS assessment discussed in Section 1.1. In addition, validation calculations of heat transfer in fuel rod bundles are performed by using the single phase solver of OpenFOAM.

2.3.1 Background

In the SAFIR2014 programme, VTT coordinated the NuFoam project, where OpenFOAM solvers were developed and validated for NRS assessment. In addition to VTT, the Aalto University, the Lappeenranta University of Technology and Fortum participated in the project that started in 2010. The project focused on the modelling of heat transfer and boiling in PWR fuel rod bundles. So far, benchmark calculations have been performed with the OpenFOAM single phase solver in fuel rod bundles. A two-phase model for heat transfer and subcooled nucleate boiling has been implemented and validated and models of direct-contact condensation have been implemented. The long term goal is to develop a general purpose two-phase CFD solver for NRS assessment in cooperation with international partners.

In commercial CFD software, the source code and the implementation of the numerical methods are not openly available. Thus, the possibilities to modify the solver or to include new models are limited. In addition, the license policy prevents effective utilization of parallel computer resources. The use of open source software in the nuclear safety analysis would increase its transparency and improve opportunities for co-operation as everyone would have access to a common software platform. The cost of parallel computational capacity has continuously been decreasing and the lack of licensing fees would allow this capacity to be utilized more effectively. This would improve the accuracy and reliability of modelling and increase the number of situations where CFD methods can be utilized.

The use of open source CFD software, especially OpenFOAM, is becoming more popular in industrial applications. In Finland, OpenFOAM is used at technical universities and it has been applied in cases brought up by the industry. In the previous SAFIR2014 program, OpenFOAM was found to be a viable platform for nuclear safety applications and a solver capable of simulation subcooled nucleate boiling was developed. In Sweden, an OpenFOAM solver for boiling in fuel rod bundles has been developed at KTH in cooperation with VTT. The work at KTH has been funded by NORTHNET Roadmap 1. The work at VTT and KTH has raised interest of several international organizations in the field of nuclear safety. Helmholtz-Zentrum Dresden-Rossendorf (HZDR) started its OpenFOAM project in 2013. HZDR and VTT have joined OpenFOAM Process Engineering Consortium that co-ordinates development funding provided by consortium partners to the OpenFOAM Foundation.

In 2014–2016, VTT has directly commissioned enhancements to the OpenFOAM Foundation release in co-operation with Iowa State University. Close cooperation with the OpenFOAM Foundation makes possible to in-
include the basic functionality needed in two-phase NRS application into the official releases of OpenFOAM. This reduces significantly the maintenance work of the implemented submodels, when new releases of OpenFOAM are published. According to the experiences obtained during the SAFIR2014 programme, inclusion of the main features of the developed models in the official releases is important for cost-effective progress in model development.

Accurate and efficient communication of the development needs and a common multiphase solver platform are keys to effective international co-operation and efficient use of the resources. Therefore, development lines that diverge in structure from the official OpenFOAM Foundation release should be avoided. On the other hand, application specific closure models and similar simulations tools can be efficiently maintained nationally as long as they are compatible with the structure and interfaces of the OpenFOAM Foundation release.

Work Package 3 consists of four Tasks: (T1) Development and validation of boiling models for NRS assessment in co-operation with OpenFOAM Foundation and international partners, (T2) Development of coupling with system codes, (T3) Development and validation of models for Direct-Contact Condensation (DCC), (T4) Heat transfer in nuclear reactor fuel rod bundle, which is an in-kind contribution of Fortum.

Partners and person months allocated to WP3 in 2017 are given in the table:

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>4.5</td>
</tr>
<tr>
<td>LUT</td>
<td>4.5</td>
</tr>
</tbody>
</table>

### 2.3.2 Task 1 (T3.1) Development and validation of boiling model for NRS assessment (VTT)

In 2015, the boiling and condensation models previously developed in the SAFIR programme were included in the multiphase solvers of the official OpenFOAM release in co-operation with the OpenFOAM Foundation. The goal was to avoid excessive maintenance of a separate solver code that structurally differs from the official OpenFOAM release. Even more importantly, the commonly available code allows efficient co-operation with OpenFOAM developers and other international partners. This has already resulted in improved solver performance and usability.

In 2016, the wall boiling models were extended to high void fractions and a framework was implemented for runtime selectable heat flux partitioning and wall boiling submodels. The boiling and condensation models were coupled to the Interfacial Area Transport Equation model (IATE) that allows modelling of bubble coalescence and break-up. Validation simulations of the implemented models have been performed by the end of the year.

As the void fraction increases, the coalescence of bubbles leads to formation of larger steam bubbles. As the bubble size increases, the lift force caused by velocity gradient of liquid is reversed. Thus large and small bubbles segregate and their transport needs to be modeled separately.

In 2017, the thermal phase change models will be extended from two-phase to multiphase solver in order to allow simulation of boiling flows with wide bubble size distribution. The coalescence and breakup framework will be extended to multiphase solver. The goal is to couple the phase change models with inhomogeneous class method (termed iMUSIG). The work will be carried out in co-operation with OpenFOAM Foundation, CFD Direct and Helmholtz Zentrum Dresden-Rossendorf (HZDR). HZDR intends to contribute their inhomogeneous class method implementation while VTT will work on extension of the thermal phase change models to multiphase and coupling of those to the inhomogeneous class method model. CFD Direct on behalf of the OpenFOAM Foundation will coordinate the integration of the models to the official OpenFOAM release.

In 2018, the polydisperse subcooled boiling toolset that will be integrated into the official OpenFOAM release in the present project will be validated and the possible identified weaknesses will be addressed. The Groeneveld look-up tables for critical heat flux and the DNB experiments of the NUPEC PSBT benchmark will be used for the validation.

This task will be performed in co-operation with HZDR, KTH and NORTHEMNET Roadmap 1. At KTH, experiments will be performed and OpenFOAM models will be developed for dry-out in BWRs. The boiling capability now integrated into the official OpenFOAM release will provide a common development platform and the aim is to increase international co-operation in the development and validation of OpenFOAM based simulation tools for NRS applications.
2.3.3 Task 2 (T3.2) Development of coupling with system codes (VTT)

Coupling of OpenFOAM with system codes will be developed. Earlier, coupling of Apros with ANSYS Fluent has been developed in co-operation of VTT and Fortum in a separate project. Implementation of similar coupling interface in OpenFOAM as has been implemented in Fluent is proposed. This would make possible the coupling of OpenFOAM with Apros and SMABRE by utilizing already existing coupling features of the system codes.

In 2017, interfaces for the most important couplings will be implemented in OpenFOAM. These include couplings at

- mass-flow-inlets
- pressure-outlets

The implementation will be verified by performing test simulation with a simplified model of pressure vessel, where the primary loops will be modelled with Apros and the pressure vessel with OpenFOAM.

In 2018, the following couplings with system codes will be implemented in OpenFOAM:

- heat transfer on walls
- volume coupling in “porous zone”

“Porous zone” means here, for instance, tube bank of steam generator or fuel rod bundle that is not described in detail in the CFD mesh. Instead, pressure loss and heat transfer are described with source terms of momentum and enthalpy. The volume coupling will be tested with a simple model of PWR core.

2.3.4 Task 3 (T3.3) Development and validation of models for Direct-Contact-Condensation (LUT)

The stability of OpenFOAM two-phase solver reached the level that allowed first simulations of chugging condensation mode during the NuFoam project of SAFIR2014 programme.

The applicability of updated compressible two-phase solver of OpenFOAM has been tested for the simulations of chugging condensation mode. During 2015, both 2D and 3D simulations of POOLEX experiment (STB-28-4) of chugging have been done with compressible two-phase solver of OpenFOAM. The preliminary analysis yielded that the averaged condensation rates of OpenFOAM simulations are comparable to the previous NEPTUNE_CFD results of Tanskanen et al. [15, 16]. However, some qualitative differences existed.

During year 2016, 2D and 3D simulations of PPOOLEX DCC-05 experiment were continued. Grid sensitivity study of the STB-28-4 was finalized. The OpenFOAM results were compared with the corresponding NEPTUNE_CFD simulations. Differences between the applied k-epsilon turbulence models in NEPTUNE_CFD and OpenFOAM cases caused notable overestimation of condensation rate in some OpenFOAM cases. The Rayleigh-Taylor Instability (RTI) model of Pellegrini [17] was implemented to the NEPTUNE_CFD solver within INSTAB project in 2015. During 2016, same model was implemented to the OpenFOAM two-phase solver and testing of it was started with the DCC-05 case.

The sparger tests of LUT are also a part of GOTHIC EHS/EMS model validation work of KTH. CFD simulations of these experiments may support both the EMS/EHS validation and the OpenFOAM development. In 2017, remaining RTI model testing efforts will be finished and simulations of a sparger will be started. As far as project authors know, a small-scale separate effect test facility will be designed and constructed at LUT in INSTAB 2017 project. In the test facility, it will be possible to measure directly the effective momentum induced by steam injection through a single orifice for different condensation regimes. This facility would be the best reference system for the sparger CFD studies, and the planned design of it would be used as the starting point of the simulations. The OpenFOAM simulations of any sparger test require changes to the modelling approach used in the straight blowdown pipe studies, but this work can be done mostly parallel with the design process of the facility during 2017 (see also Chapter 6: Risk Management). Sparger simulations will continue during 2018.

Due to the evaluation process of SAFIR applications, CFD simulations included in INSTAB 2017 project were moved to NURESA, and their funding was decreased. The goal in these INSTAB simulations was to exploit the extensive database gathered in the previous POOLEX studies of steam discharge into a pool of sub-cooled water in the assessment of the capability of CFD codes to simulate direct-contact condensation situations. In INSTAB 2016, a plexiglass blowdown pipe experiment done in POOLEX (TRA-4) was simulated with 2D hexahedral mesh with NEPTUNE_CFD. Numerical problems were worse than, e.g., in the smaller pipe case (PPOOLEX DCC-05) simulated previously. These problems were overcome just at the end of the project, and new information of the simulations and analysis of experimental results in larger blowdown pipe cases were obtained. In 2017, a test done with the collar shaped outlet (POOLEX COL-series) of the blowdown pipe will be simulated with CFD. In 2018, sparger simulations were planned also in the INSTAB task. Obtained funding is
insufficient for completing these tasks, however. Both the studies, originally planned sparger case and later added collar case will be started during 2017, but only interim status will be reported at the end 2017. Results of both cases will be reported in the final report at the end of 2018.

2.3.5 Task 4 (T3.4) Heat transfer in nuclear reactor fuel rod bundle (in-kind contribution of Fortum)

In 2015–2016, flow and heat transfer in Loviisa NPP fuel rod bundle has been calculated with a single phase solver of OpenFOAM. The aim has been to update the current fuel rod bundle CFD model to better performance and accuracy. Heat transfer calculations with the updated model will be finished by the end of year 2016. A special interest has been in the evaluation of workflow efficiency.

In 2017, no work will be done in this Task.
This Task has been an in-kind contribution of Fortum to the project and it has been fully funded by Fortum.

2.4 Work Package 4 (WP4) Coupled CFD-Apros simulations of NPP components

Coupling of Apros and CFD code enables detailed three-dimensional modelling of one process component that is connected with a complicated system of pipelines and other process components. Two different types of coupling can be readily identified. In one-way coupling, Apros simulation provides boundary conditions for the CFD calculation, but no feedback from the CFD calculation to the Apros simulation occurs. In two-way coupling, the CFD code also provides boundary condition for the Apros simulation.

In Apros 6, two-way coupling of Apros with ANSYS Fluent CFD code has been implemented at VTT in cooperation with Fortum in a separate project. Numerical stability of the co-simulation has been an issue. It is necessary that the codes exchange information several times within each time step. Semi-implicit coupling of the codes has been found to be a suitable method.

In Work Package 4, verification and validation calculations for different Fluent-Apros coupling scenarios are performed. This Work Package belongs to the topic “Coupling with third party codes” in the Roadmap presented in Section 1.1.

2.4.1 Two-way coupled CFD-Apros simulations

Two-way coupled CFD-Apros simulations of NPP components of the primary circuit of PWR are performed. The most interesting components for three-dimensional CFD simulation are the steam generator, pressurizer and the pressure vessel. In 2016, two-way coupled Fluent-Apros simulation of a steam generator is being performed, where generic Apros model of VVER-440 plant is coupled with CFD model of steam generator.

In 2017–2018, simulation of a pressurizer is proposed, where generic model of VVER-440 plant will be coupled with CFD model of the pressurizer. In addition, submodels will be developed for the condensation of steam on spray droplets and interaction of wall film with droplets and vapor. Suitable test cases for the validation simulations will be chosen in cooperation with the Reference Group.

The CFD model for the pressurizer will be coupled with the plant model at the surge line, spray injection, heater element and possibly measurements of surface level or pressure.

Partners and person months allocated to WP4 in year 2017 are given in the table:

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>3.3</td>
</tr>
<tr>
<td>Fortum</td>
<td>In-kind contribution (1 person month)</td>
</tr>
</tbody>
</table>

2.4.2 Task 1 (T4.1) Development of two-way coupled CFD-Apros model of a NPP pressurizer (VTT)

In 2017, CFD model of a VVER-440 pressurizer will be constructed by Fortum in Task 4.2 for the ANSYS Fluent CFD code. The model will be used in the development of the physics models needed for the simulations of pressurizer.

First, the most important heat transfer models will be implemented:
• Heat transfer model for the heater elements including model for boiling
• Bulk evaporation and condensation models
• Interfacial condensation and evaporation on the water surface

In addition, the steam tables will be adapted into the pressurizer model. The models developed earlier for steam generators will be used.

Models for the evaporation and condensation of spray droplets will be adapted to the pressurizer model. Previously, evaporation and condensation models of droplets have been developed for the single phase solver of Fluent in the presence of non-condensable gas. Lagrangian description of the droplets will be used, where the trajectories of the droplets are resolved in the numerical mesh.

The evaporation and condensation models of droplets will be modified for the Euler-Euler multiphase solver of Fluent for the situations, where non-condensable gas is not present. The diffusion based condensation model will be replaced by pressure based condensation model.

Test calculation of pressure rise or pressure reduction will be performed for a VVER-440 pressurizer.

![Figure 5. Schematic presentation of VVER-440 pressurizer (Takasuo, 2006).](image)

### 2.4.3 Task 2 (T4.2) Evaporation and condensation of thin liquid films on the walls (VTT)

In 2018, testing and validation of the spray model will be continued. In addition, the model of thin liquid films on the walls will be adapted in the pressurizer model.

Eulerian model for thin liquid films will be used for the modelling of wall condensation and evaporation. The film model has earlier been tested for the modelling of condensation at VTT. The model assumes laminar velocity profile in the film and solves the thickness and the average velocity of the film. In addition, the interaction of the film with liquid droplets will be included in the model.

The evaporation model will be implemented in the film model. The pressure based evaporation model will be used for the case, where non-condensable gas is not present. Models previously developed in the SAFIR2014 Programme will be utilized.
2.4.4 Task 3 (T4.3) Verification of CFD-Apros model of a NPP pressurizer (in-kind contribution of Fortum)

In 2017, Fortum will build the geometrical model and preliminary numerical mesh for the VVER-440 pressurizer. In addition, test simulations with the pressurizer model will be made by using physical models developed earlier and by using physical models developed during 2017 in Task 4.1. The simulations will first be made without coupling to the VVER-440 plant model. Test simulations consist of operational condition and one simple transient (to be decided).

In 2018 test simulations with coupled Apros-CFD model with updated physical models will be made.

Work will be in-kind contribution of Fortum to the project and fully funded by Fortum. Estimate for the work required in 2017 is 1.0 person months.
2.5 Work Package 5 (WP5) Coordination and international co-operation

The project is managed and coordinated at VTT by Dr Timo Pättikangas, who is certified project manager (International Project Management Association, IPMA Level C certificate).

The progress of the Work Packages is monitored in project meetings and in SAFIR2018 Reference Group 4 meetings.

In addition to the Reference Group 4, the work done in WP2 is reported in the NORTHNET Roadmap 3 Reference Group meetings and to the NKS Steering Committee.

The work done in Work Package 3 is also reported in the NORTHNET Roadmap 1 meetings.

In the development of the OpenFOAM two-phase models on boiling, co-operation with the research group of prof. Henryk Anglart at KTH and Roland Rzehak at the CFD group at HZDR is done. In addition, development work is done in close co-operation with the OpenFOAM Foundation, CFD Direct Ltd and the international OpenFOAM Process Engineering Consortium.

In 2017, participation in the Northnet Roadmap 1 and 3 meetings will be reduced.

Partners and person months allocated to WP5 in year 2017 are given in the table:

<table>
<thead>
<tr>
<th>Partners in WP5</th>
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</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
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</tbody>
</table>
## 3. Milestones in October 2017

The milestones of the NURESA project that are to be achieved by October 2017 are listed in the table:

<table>
<thead>
<tr>
<th>Milestone number</th>
<th>Milestone</th>
</tr>
</thead>
<tbody>
<tr>
<td>M2.1.1</td>
<td>Pre-calculation of small-scale separate effect test on condensation has been performed. Draft report has been written. (VTT)</td>
</tr>
<tr>
<td>M2.2.1</td>
<td>The first CFD calculations of the spray effect on the stratification in the wet well have been performed. (VTT)</td>
</tr>
<tr>
<td>M3.1.1</td>
<td>Thermal phase change modelling capability has been implemented in OpenFOAM multiphase solver. (VTT)</td>
</tr>
<tr>
<td>M3.1.2</td>
<td>Polydisperse subcooled nucleate boiling simulations with OpenFOAM-dev are in progress. (VTT)</td>
</tr>
<tr>
<td>M3.2.1</td>
<td>Coupling of OpenFOAM with system codes has been implemented. Draft report has been written. (VTT)</td>
</tr>
<tr>
<td>M3.3.1</td>
<td>Direct-contact condensation simulations with OpenFOAM by using the Pellegrini Rayleigh-Taylor instability model have been finalized. (LUT)</td>
</tr>
<tr>
<td>M3.3.2</td>
<td>Geometry of separate effect sparger test system built-up and OpenFOAM pre-simulations are underway. (LUT)</td>
</tr>
<tr>
<td>M4.1.1</td>
<td>CFD model for the pressurizer is ready. First test calculations have been performed. (VTT)</td>
</tr>
</tbody>
</table>
## 4. Deliverables 2017

The planned deliverables and responsible organizations and persons for 2017 are listed in the table:

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name and description</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D2.1.1</td>
<td>Report on the CFD calculations of the condensation in the small-scale separate effect facility. (VTT)</td>
<td>2.0</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D2.2.1</td>
<td>Report on the CFD calculations of the spray effect on the stratification in the wet well. (VTT)</td>
<td>2.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Thermal phase change capability in OpenFOAM-dev reactingMultiPhaseEulerFoam. (VTT)</td>
<td>1.9</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D3.1.2</td>
<td>Report on polydisperse subcooled nucleate boiling simulations with OpenFOAM-dev. (VTT)</td>
<td>1.6</td>
<td>31.12.2017</td>
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<tr>
<td>D3.2.1</td>
<td>Short report on the coupling of OpenFOAM with system codes. (VTT)</td>
<td>1.0</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D3.3.1</td>
<td>RTI model implementation to OpenFOAM and interim status of single orifice sparger and collar blowdown pipe simulations. (LUT)</td>
<td>4.5</td>
<td>30.1.2018</td>
</tr>
<tr>
<td>D4.1.1</td>
<td>Report on the CFD model for a VVER-440 pressurizer. (VTT)</td>
<td>3.3</td>
<td>31.12.2017</td>
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<td>In-kind contribution</td>
<td>15.1.2018</td>
</tr>
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<td>D5.1.1</td>
<td>Coordination of the project; participation in Northnet meetings, co-operation with KTH, HZDR and CFD Direct. (VTT)</td>
<td>0.4</td>
<td>31.1.2018</td>
</tr>
</tbody>
</table>

| Total pm           | 17.2                                                 |
5. Project organization

The project manager of the NURESA project is Dr Timo Pättikangas at VTT Technical Research Centre of Finland. VTT is the responsible organization for the project. The project manager is responsible of the execution of the project according to the NURESA project plan and guidance given by the Reference Group of the project. The deputy project manager is M.Sc. Juho Peltola at VTT.

Each Work Package has WP leader that is responsible of the work done in the work package:

- WP1: M.Sc. Ville Hovi, VTT
- WP2: Dr Timo Pättikangas, VTT
- WP3: M.Sc. Juho Peltola, VTT
- WP4: Dr Timo Pättikangas, VTT
- WP5: Dr Timo Pättikangas, VTT

All Work Packages except WP3 and WP4 are fully performed at VTT. WP3 is a joint activity of VTT, LUT and Fortum. The responsible persons of the Tasks of WP3 are the following:

- T3.1: M.Sc. Juho Peltola, VTT
- T3.2: M.Sc. Ville Hovi, VTT
- T3.3: Dr Vesa Tanskanen, LUT
- T3.4: M.Sc. Timo Toppila, Fortum

Note that Task T3.4 is an in-kind contribution of Fortum, where no work is done in year 2017.

WP4 is a joint activity of VTT and Fortum. The responsible persons of the Tasks of WP4 are the following:

- T4.1: Dr Timo Pättikangas, VTT
- T4.2: Dr Timo Pättikangas, VTT
- T4.3: M.Sc. Timo Toppila, Fortum

The progress of the project is reported according to the rules and practices of the SAFIR2018 programme.

The main researchers, their organization, the tasks they will be contributing, and the estimated person months in 2017 are listed in the following table:

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organization</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Timo Pättikangas</td>
<td>Principal Scientist</td>
<td>VTT</td>
<td>T2.1, T2.2, T3.1, T4.1, T5.1</td>
<td>5.0</td>
</tr>
<tr>
<td>Juho Peltola</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T3.1, T3.2</td>
<td>3.5</td>
</tr>
<tr>
<td>Ville Hovi</td>
<td>Research Scientist</td>
<td>VTT</td>
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<td>3.5</td>
</tr>
<tr>
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<td>VTT</td>
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</tr>
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<td>Risto Huhtanen</td>
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<td>VTT</td>
<td>T2.2</td>
<td>0.4</td>
</tr>
<tr>
<td>Vesa Tanskanen</td>
<td>Dr</td>
<td>LUT</td>
<td>T3.3</td>
<td>1.0</td>
</tr>
<tr>
<td>Giteshkumar Patel</td>
<td>M.Sc.</td>
<td>LUT</td>
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</tr>
<tr>
<td>Tommi Rämä</td>
<td>M.Sc.</td>
<td>Fortum</td>
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<td>In-kind contribution</td>
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<tr>
<td>Timo Toppila</td>
<td>M.Sc.</td>
<td>Fortum</td>
<td>T4.3</td>
<td>In-kind contribution</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>17.2</strong></td>
</tr>
</tbody>
</table>
6. Risk management

The main risk types in the NURES project are:

- Technical risks related to simulation software
- Time table risks
- Availability of the key personnel

In WP2, spray experiments of PPOOLEX facility are modelled. The main risk is the time schedule of the availability of the experimental data. If the design of the small-scale separate effect facility of condensation is delayed, the data on the experimental geometry and experimental parameters may not be available. This would delay the pre-calculations. The time table risks are reduced via close co-operation with the experimental group at LUT.

In WP2, the desired well instrumented single orifice separate effect test facility is assumed to be funded within SAFIR INSTAB project. If funding is not obtained, another single orifice test has to be chosen as a reference. Other geometries are more complicated and less measurement information is available, which may decrease the usability of the simulation results.

In WP3, OpenFOAM models are developed and validated. HZDR intends to contribute their inhomogeneous class method implementation to the official release of OpenFOAM. The discussions on the terms of the contribution are still pending and could cause changes to the plans of WP3.

In WP3, the availability of the key personnel is also important because the number of experienced OpenFOAM users is currently too small. One of the goals of WP3 is, however, to train new OpenFOAM experts.

In WP4, coupled CFD-Apros simulations of pressurizer are proposed. Several physics models on sprays, wall films, steam tables and coupling with Apros need to be implemented. Technical problems may cause delays in the progress of work package. The researchers working in this WP are, however, quite experienced with similar problems, which reduces the risks.
References


### Development and validation of CFD methods for nuclear reactor safety assessment

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Volume</td>
<td>Personnel</td>
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<tr>
<td>WP1 - CFD benchmarks</td>
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<td>T3.3 Direct-contact condensation (LUT)</td>
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<tr>
<td>WP4 - Coupled CFD-Apros simulations</td>
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<td>43</td>
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<tr>
<td>T4.1 Two-way coupled simulation of pressurizer (VTT)</td>
<td>3.3</td>
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<tr>
<td>T4.3 Verification of the model of pressurizer (Fortum, in-kind)</td>
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<tr>
<td>WP5 - Project coordination</td>
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<tr>
<td>T5.1 Coordination</td>
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<td><strong>TOTAL</strong></td>
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### Comments:
- Other costs in WP1, WP2, WP3 and WP4 are the fees of the computing environment at VTT.
- In-kind contribution of Fortum is included in WP4 Task 4.3, where Fortum participates the modeling of NPP pressurizer. The volume is about one person month.
- External service: In WP3, service from OpenFOAM Foundation for the inclusion of required features in the OpenFOAM multiphase solver.
The Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2018)

Resource Plan for 2017

NURESA VTT Budget

Development and validation of CFD methods for nuclear reactor safety assessment

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</tr>
<tr>
<td>WP1 - CFD benchmarks</td>
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<td>T1.1 ROCOM experiment PKLIIIT1.1 on mixing</td>
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<td>1,0</td>
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<td>3,3</td>
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<tr>
<td>T4.1 Two-way coupled simulation of pressurizer</td>
<td>3,3</td>
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<td>WP5 - Project coordination</td>
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Comments:
Other costs in WP1, WP2, WP3 and WP4 are fees of the computing environment at VTT.
In-kind contribution of Fortum is included in WP4, where Fortum participates by computing a transient in an NPP pressurizer. The volume is about one person month.
External service: In WP3, service from OpenFOAM Foundation for the inclusion of required features in the OpenFOAM multiphase solver.
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<tr>
<td>TOTAL</td>
<td>4,5</td>
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</table>

Comments:
SAFIR2018 Project plan

PANCHO

Physics and Chemistry of Nuclear Fuel

Ville Tulkki, Asko Arkoma, Timo Ikonen, Joonas Kättö, Henri Loukusa, Emmi Myllykylä, Rami Pohja
VTT Technical Research Centre of Finland ltd
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1. Research theme and motivation

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 the chemistry of the fuel pellet and the mechanical response of the cladding.

SAFIR2018 Update to the Framework plan for 2017 call states:

There will be a change in the electricity production modes supporting the grid. The production that is dependent on weather conditions will increase and at the same time the production based on fossil fuel will decrease. Less conventional base load capacity will be available for maintaining the power balance and this may also have effects on the operation of the nuclear power plants. Potential safety issues related to the operation of nuclear power plants in the load following mode are an important topic for research.

and impresses the importance of the research on the effects of load follow on nuclear fuel. For RG2 topics, this is further elaborated:

Research on the effects of load follow on nuclear fuel and pressure boundary integrity (fatigue due to varying loads) is important. It would be useful to review how the fuel models used in Finland can be applied in the analysis of fuel loading in the load following operation of the plant, and how the Finnish plants would respond to load follow demands

The importance of this research is further emphasized by the events in the spring 2016 in Olkiluoto, where several fuel rods failed, potentially due to plant operation related stress corrosion cracking. During the year 2017, these issues are focused on in the Task 3 of PANCHO.

1.1 Background and state-of-the-art

1.1.1 Integral fuel behaviour

The nuclear fuel behaviour during reactor operation and accidents is commonly modelled using so-called integral fuel codes that use models and correlations derived from experiments. These models are often empirical, and therefore understanding the domain of validity of the codes is of great importance. The models are often validated for a certain set of conditions, with different models used for describing behaviour at different conditions. The selection of the models used is often based on the expert judgement of the user of the code, requiring the understanding of the phenomena and tools from the users of the codes
also. Currently the codes used in Finland have been developed abroad, examples being British ENIGMA, French SCANAIR, and FRAPCON and FRAPTRAN from the United States of America. Finnish work has been done to most of the codes to better model the domestic nuclear power plants.

The code validation is performed against large databases of experimental data. The experiments on nuclear fuel are complex and expensive. Therefore they require international consortia which are vital for the transfer of information and expertise. The current European experimental facilities include Halden Reactor in Norway. The CABRI reactor has been reworked to facilitate experiments in a water loop and is nearing operational condition, while the future Jules Horowitz Reactor is being built in France.

As previously noted the models currently in use in integral fuel behaviour codes rely on correlations tuned to experimental results. This limits the applicability of the codes to materials and conditions previously analysed, and hinders the adoption of new solutions and materials. As it is, there is a growing need for a more mechanistic approach in the analysis methods, and such more scientific description needs to be integrated with the current engineering level practices and codes. However, full understanding of the fuel behaviour phenomena is only developing.

Most of the fuel codes are designed to be stand-alone codes taking input from external source without two-way feedback to for instance neutronics or thermal hydraulics codes. A relatively new field of study called multiphysics analysis is combining codes from various fields in an effort to be able to better predict the interaction of various processes. This is facilitated by increasing calculational capabilities. However, the forced coupling between the various dedicated codes originally meant for stand-alone analysis has often lead to practical problems. A new fuel code developed at VTT, FINIX, is aimed especially for multiphysics simulations, where it takes the role of the simulation’s fuel behaviour model. FINIX has been designed to be integrated into a wide array of simulation codes, and to provide an identical description of the fuel thermal behaviour across different disciplines such as reactor physics and thermal hydraulics. The FINIX approach is to be as simple as possible in order to cater to a wide audience, in contrast to most of the multiphysics codes that strive for information in utmost detail from all possible sources to perform mechanistic predictions (e.g. BISON).

For safety analysis of the nuclear fuel behaviour, understanding the accident behaviour is of paramount interest. The accidents that the fuel may experience fall roughly into two categories: loss of coolant accidents and reactivity initiated accidents.

A large-break loss of coolant accidents (LB-LOCAs) in a water-cooled reactor consist of a break of the coolant primary system and the consequent loss of core cooling capacity. LOCA is a design basis accident, that is, a type of accident that the nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety. This means that safety authorities generally require that LOCA analyses are performed to ensure that the plants would survive these accidents without major release of radioactivity. Consequently, a great deal of research has been conducted all over the world to study the fuel behaviour under LOCA conditions.

The past studies have provided the basis of the present LOCA acceptance criteria and procedures practiced by the nuclear industry and authorities. However, new fuel rod designs are being constantly developed as there seems to be a need to expose fuel rods to increasingly demanding operating environments due to financial pressures. For example, the traditional Zircaloy-4 claddings in PWRs, for which the most past LOCA tests were made on, are being replaced by Zr-Nb base alloys because of their better performance in high burn-up conditions. As previous LOCA research has shown strong alloy composition and burn-up effects, there is a need to evaluate the LOCA behaviour of new fuel designs after varying irradiation histories. The need for improved understanding of the fuel behaviour during LOCA after the Fukushima accident is also evident in the initiation of the IAEA’s new accident focused coordinated research project FUMAC.

A reactivity initiated accident (RIA) involves a fast increase in reactor power. This power increase may result in a severe damage the reactor core, and some accident scenarios for RIA have been identified by regulatory bodies as design basis accidents. It is important to maintain and further develop RIA modelling
capabilities for all types of reactors in Finland (EPR, BWR, VVER). From the development point of view, especially the post-DNB behaviour of the rod during an RIA is not well known, and therefore the modelling of that phase is not yet so advanced. In past and ongoing experimental projects, the coolant conditions have not been representative of those in LWRs. To fix this deficiency, the international experimental RIA research programme CABRI International Programme (CIP) is to investigate the RIA behaviour in a new water test loop. It would produce necessary information to qualify the RIA modelling codes. For instance, SCANAIR used at VTT has a 1-dimensional single-phase model for thermal hydraulics which could be improved using the experimental data. Also, SCANAIR lacks a thermal hydraulics model for BWR conditions, and the development of such modelling capabilities is a special subject to address in the Finnish context.

Following the recommendation from the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) technical workshop titled “Nuclear Fuel Behaviour during Reactivity Initiated accidents” (2009), a RIA fuel codes benchmark was organized between 2011–2013 under the Working Group on Fuel Safety (WGFS). In the simulated cases, when using the given specifications to build the code inputs, various codes produced different results compared to each other especially with respect to the cladding temperatures and cladding hoop strains. The scatter was largest in the cases in which water boiling occurred. Therefore, the recommendation from the first benchmark was to launch a second phase exercise in which the emphasis would be put on deeper understanding of the differences in modelling of the different codes; in particular, looking for simpler cases than those used in the first benchmark. In addition to this first activity, a second activity was planned in which an assessment of the uncertainty of the results in various codes is made based on a well-established and shared methodology. The first activity of the second phase of the benchmark started in 2014, and the second activity started in 2015 and ended in 2016. In the second activity, various influential input parameters were identified for fresh fuel. A new, third phase for the benchmark is now proposed, starting in 2017. In the new phase, the focus would be on uncertainty and sensitivity analyses on an irradiated case, in order to identify the corresponding influential input parameters. This information would be useful in establishing a Phenomena Identification and Ranking Table (PIRT) for RIA and could guide the future RIA tests and code improvements.

It has also been demonstrated that the proper understanding of the accident behaviour relies on the ability to understand the state of the nuclear fuel evolved during the base irradiation. The current research issues include the effects of new cladding and fuel pellet designs, the behaviour of fission gases, and the formation of gas bubble rich regions and high burnup structure. These will affect the fuel behaviour during both LOCA and RIA and should be well characterized.

In PANCHO, the Finnish fuel code FINIX is developed and validated for simulations of the fuel behaviour across a wide range of scenarios, such as loss of coolant accidents and reactivity insertions. FINIX can be then implemented in a wide range of codes to provide systematic description of fuel behaviour. This simplified approach to multiphysics is unique in the world. The tools and expertise to analyse reactor safety during loss of coolant and reactivity initiated accidents in Finnish reactors are improved via strong international co-operation. Phenomena pertaining to LOCA and RIA are studied and the understanding is transferred to Finnish experts and tools.

### 1.1.2 Pellet chemistry

The influence of chemical processes on the in-pile behaviour of nuclear fuel is not very well understood. With increasing burnup, various chemical phenomena take place in the fuel rod, such as formation of separate solid phases from an initially homogeneous material. These chemical processes influence not only the material properties of the fuel but also of the cladding via gas phase diffusion and especially during hard contact between the pellet and the cladding. The possibility of damage to fuel due to pellet-cladding interaction is a complex process involving both mechanical stress and active chemical processes.
Additionally, in case of a defective fuel rod or in accidental conditions, the chemical properties determine the release behaviour of radionuclides from the fuel.

The chemical composition of nuclear fuel is very complex, and Gibbs energy minimization (GEM) provides a tool with which such complex systems can be studied [3]. GEM is commonly used in industry and research worldwide. Only in recent times has the increase in computing power brought about the possibility of applying these principles to such chemically complex systems as nuclear fuel. The chemical state of nuclear fuel [4], iodine stress corrosion cracking [5] and fission product release [6] have been studied with GEM. The fact that GEM relies on basic thermodynamic principles makes it possible to easily model the chemistry of different types of fuels, such as those containing additives. Such phenomena as iodine stress corrosion cracking, cladding lift-off and various effects of burnup on fuel material properties are influenced by chemical processes in the fuel.

A GEM routine has been previously developed at VTT [3] and in this project it will be used to investigate dissolution phenomena in the reactor or in the spent fuel storage pool. The oxidation state of fuel affects the material properties of fuel and also fission product release from fuel. Both of these processes are important when describing the in-pile performance of defective fuel.

In the case of defective fuel, liquid water may come into contact with the fuel, oxidizing the fuel pellet and dissolving some of its surface. Some of the chemical processes in fuel can be experimentally studied with analogue materials, non-irradiated UO2 fuel or with simulated fuel SIMFUEL. During the previous decades, different kinds of dissolution experiments have been conducted with fuel materials as function of pH or salinity mainly in the context of disposal studies of spent nuclear fuel. These experiments have been conducted either under atmospheric or anoxic (reducing) conditions. In some cases the radiation effects have been taken into account in some form. Some leaching studies under reactor water conditions can be also found in literature, but these studies are not very extensive. In LWR’s water is as pure as possible and the water conditions are normally maintained reducing and pH near neutral to hinder the corrosion of reactor materials. [7] Some components (Zn, H, noble metals) may be added into water or on metal surfaces to create the most favourable conditions considering the minimization of material degradation, fuel performance issues and the control of the radiation field. Depending how the defective fuel bundles are stored, the conditions can turn into more oxidising ones if the spent fuel storage pool is in contact with the atmosphere [8]. The coolant water of the spent fuel storage pool can also be more acidic than that in the reactor due to added boric acid.

CeO2 and ThO2 have been used as structural analogues for the fluorite-type structure (space group Fm3m) in UO2 dissolution studies [9]. Thorium occurs naturally α-active like uranium. However, unlike U(IV)O2, Th(IV)O2 is not redox-active since Th has only one prevailing oxidation state, +4. Next generation applications of nuclear energy have also shown interest towards thorium [10]. As a fuel, thorium has many beneficial properties, such as high fusion temperature, good sintering capability, resistance against radiation damage, greater abundance in the Earth’s crust compared to U, and the possibility for transmutation [10]. Simulated fuels contain elements which mimic the behaviour of different fission products in fuel matrix [11]. The results of the SIMFUEL leaching experiments increase not only knowledge of the behaviour UO2 matrix, but also the behaviour of fission products.

1.1.3 Cladding mechanical response

In engineering applications the creep behavior of fuel cladding tube materials, such as zircalloys, are usually described consisting of the stages of primary, secondary and tertiary creep regions. The newly developed creep relaxation model logistic creep strain prediction (LCSP) [16 - 19] attempts to describe the creep behaviour of these stages. However, a universal challenge with the majority of creep strain models, including LCSP, is the ability to characterize the early stage of creep deformation. This stage is commonly referred to as the primary creep stage or transient creep. The challenges are both in determining the initial strain evolution and the creep response to changing conditions. The latter is very relevant in fuel behaviour analysis due to stresses caused by pellet cladding mechanical interaction. Furthermore, the material
conditions affect significantly the creep behaviour fuel claddings made of zirconium-based alloys. The chemical composition, the material condition (e.g. stress relieved, recrystallized) have effect on the material creep behaviour and thus they should be taken into account in creep models.

The initial strain evolution can be studied with stress relaxation testing. Stress relaxation refers to material response whereby stress under structural constraint is decreased (relaxed) by creep. In a common experimental application, the total strain of a uniaxial test specimen is fixed and load (stress) is monitored as it is gradually reduced by conversion of elastic to inelastic strain at a constant temperature. In order to evaluate the predictability of the creep strain models both in transient creep and in the steady state creep regime, experimental data covering the very early stages of creep deformation (stress relaxation tests) as well as steady state creep (mid-term creep tests) are required.

The transient creep response of zirconium alloys is conventionally handled with the strain hardening rule, yet it is well established not to apply in many situations. A methodology based on viscoelastic properties of zirconium alloys has been under development at VTT to better describe the mechanical response to changing conditions [20, 21]. The anelastic component in the viscoelastic behaviour of the cladding material at the stage of transient creep can be carefully studied by means mechanical loss spectroscopy. Such works already been performed earlier on Zr alloy material [22], but currently the amount of published data is very limited. It is also very well known that viscoelasticity at high temperatures strongly depends on the type and the concentration of alloying elements, grain size and its orientations toward applied load [23]. PANCHO experiments will provide a very important piece of information for further development of the creep model. In case sufficient experimental data is generated, such a study may potentially enable to predict a value of expected relaxation depending on the initial composition of applied Zr-alloy and its microstructural characteristics.

In PANCHO the issue of transient creep is studied both theoretically and experimentally. The produced data combined with the more accurate creep strain models can be utilized to improve the tools used in the estimation of fuel cladding tube behaviour and lifetime.

1.2 Objectives and expected results

At the end of PANCHO, the FINIX fuel code is to be able to provide predictions of fuel LOCA and RIA behaviour both as a stand-alone fuel code and as a fuel module implemented in other codes. This will include an ability to model the steady state irradiation. The FINIX code is validated with the concurrently developed and expanded validation system. The validation system SPACE will be rewritten with the aim of being able to provide automated validation for the various fuel performance codes in use at VTT and the validation matrix is to be expanded.

Currently open questions regarding the fuel behaviour during loss of coolant accidents are regarded as the ones with most safety significance. The current understanding is to be implemented to the codes that are being developed and validated. This is to be done within the Finnish participation of IAEA Coordinated Research Project FUMAC (Fuel modelling in accident conditions). A major goal for the LOCA work is to reach the state-of-the-art knowledge of the LOCA phenomenon, which will eventually allow developing VTT’s in-house fuel performance codes to a level that will enable reliable LOCA analyses.

The organization developing SCANAIR, Institut de Radioprotection et de Sûreté Nucléaire (IRSN) has granted a licence to the SCANAIR software against VTT’s yearly in-kind contribution of work. The annually negotiated and approved contribution includes development and validation of models relevant to Finnish nuclear fleet. In addition to BWR thermal hydraulics modelling, one of the tasks is foreseen to be the verification and adding of existing and new VVER-specific material property correlations in SCANAIR.

The third phase of RIA fuel rod codes benchmark exercise organized by the OECD/NEA Working Group on Fuel Safety will be participated in. In the benchmark, the objective is to assess the ability of RIA fuel rod codes to reproduce the results from experiments and to evaluate the uncertainties associated to the values they calculate. In the second phase of the benchmark that ended in 2016, simplified fresh fuel cases were
considered. In the third phase, the same uncertainty and sensitivity analysis methods applied in phase two are adopted for irradiated fuel. The most important input parameters in irradiated fuel RIA will be distinguished.

A close collaboration with Halden Reactor Project is maintained, both by the participation in the technical work in the way of in-kind research, utilization of Halden results to own research as well as by representation in the Halden Program Group and by disseminating the information to Finnish stakeholders.

The leaching experiments of fuel analogues aim to produce data for validation of chemical behaviour of defective fuel. In the experiments, the goal is also to pay attention to phenomena taking place at the fuel surface, such as re-precipitation and formation of secondary phases, not only to the release of the elements from the fuel matrix. The experiments were started with the the analogue material ThO$_2$. This allowed the development of experimental conditions and setup towards more realistic ones in the experiments with UO$_2$ based SIMFUEL. The goal is to determine the release rates for elements leached from ThO$_2$ and SIMFUEL. An objective will be that the knowledge gathered either with experiments or models support each other as the project proceeds.

In the modelling of the behaviour of a defective fuel rod, dissolution of fuel is studied with the use of GEM. During shutdown or in interim storage, defective fuel may come into contact with liquid water, so aqueous solution modelling capabilities are needed. Experimental results from fuel leaching studies are used to obtain information on to what extent the currently available data in the literature, particularly that in the OECD/NEA Thermochemical Database project, is applicable to higher temperatures than room temperature. The results are published as conference and journal publications.

The goal of the cladding investigations is to create a physically justified and experimentally validated methodology and models to describe the cladding behaviour. Objectives and a rough workplan to support this goal are as follows. A methodology for describing cladding mechanical response to transients is to be developed and a model based on it is formulated. A doctoral thesis based on this work is completed. Available experimental capability is surveyed. Potential suppliers for the test samples are contacted. A test series is designed using available tools and materials and executed during the project. The focus of PANCHO is in the transient response of the cladding and the experiments will be planned accordingly. The experimental results are used for model development and validation. The models will be implemented in both fuel behaviour codes and standalone FEM codes to be available for use in evaluation of the cladding performance.

1.3 Exploitation of the results

The FINIX code, when validated to its intended domain of use, will be able to provide feedback on fuel behaviour to wide array of simulation platforms. The design philosophy for FINIX has been the application first, and as such it has been already integrated to several reactor safety analysis codes. All the developments to FINIX will be readily usable by the end users. The development of an in-house fuel code will increase the Finnish expertise in the field substantially more than would be possible by merely using such codes. This expertise is immediately available for safety analysis work.

Understanding the accident behaviour of nuclear fuel is important for ensuring the safety of the operation of nuclear power plants. Active work on model development, validation and interpretation on LOCA and RIA relevant models provides ability to understand and quantify the safety significance of various scenarios.

As the experimental data on the fuel behaviour is the backbone of the safety analysis, participation in the Halden Reactor Project and observation of the CABRI International Project is vital. The experimental data and analyses of the results are immediately usable by the utilities, research organizations and the regulator for safety related work. As the fuel designs continually improve the on-going experimental work is required for assessments of the new designs. An innovative methodology for describing cladding creep
based on viscoelastic theory that was recently developed as a part of SAFIR2014 project PALAMA is one example of the use of Halden experimental data in SAFIR projects.

The knowledge on chemical behaviour and modelling of operational nuclear fuel gathered and developed during this project can be used to improve fuel behaviour analyses, as many phenomena in the fuel rod influenced by chemistry such as stress corrosion cracking from pellet cladding interaction are poorly understood at the moment. Additionally, expertise on the chemistry of operational nuclear fuel is currently limited in Finland. The improvement of chemical modelling capabilities during this project would bring the knowledge at VTT to an international level in this matter. The work on fuel chemistry also serves to strengthen the co-operation between the modelling and the experimental studies as well as to bridge the gap between fuel analysis in the reactor and the final repository conditions.

The results of the studies concentrating on cladding mechanical response provide information on the in-reactor performance of the fuel. The demands on the accuracy of the nuclear fuel performance analysis increase constantly as the discharge burnup is raised, new reactor operation modes such as load following are considered and new materials are introduced. Increasing the understanding on the cladding short-term behaviour can reduce the need for conservatism in the fuel conditioning related to reactor start-up. The combination of short- and long-term description of the cladding behaviour facilitates the accurate description of cladding creep during full lifetime, from reactor to final repository. Creep models can be put into immediate use in safety analysis. The methodology work, if successful, will demonstrate the improvements in the chosen approach to the cladding modelling, and will in the best-case scenario be adopted internationally.

1.4 Appropriateness of the project to SAFIR2018 programme

Ensuring and increasing expertise in the field of nuclear fuel behaviour analysis is in line with the SAFIR2018 programme goals, as the fuel behaviour analysis is one of the focus areas of reactor safety analysis. Continuing the development of a Finnish fuel behaviour code is also explicitly mentioned in the framework plan, and such work is done in PANCHO in the way of FINIX development.

As the nuclear fuel provides the first physical safety barriers against the spread of radionuclides, understanding the fuel behaviour during accidents is vital. This information must be gained through experiments, as the phenomena encountered are complex. A large part of PANCHO focuses on international programmes either performing experiments or distributing the data and tools for and experience on fuel behaviour analyses.

PANCHO investigates several nuclear fuel related issues with an interdisciplinary approach combining theoretical and experimental investigations. This work supports both the in-depth understanding of safety-related phenomena and communication across disciplines. The research will support several dissertations, and thus it will be reported in scientific journals and conferences to ensure the high quality and visibility of the work. As such the project is both relevant to nuclear safety and aims for high scientific quality.

1.5 Education of experts

PANCHO staff includes eight young generation scientists, two with a doctoral degrees and six working on one with the work performed in the project. There is a constant need for education as most of the researchers at VTT active in the field have less than 5 years of experience on fuel behaviour analysis.

The possibilities for young scientists’ extended visits in foreign research institutes, such as Halden reactor in Norway or IRSN in France, are pursued as they provide important learning experiences. The results of the research performed in PANCHO are aimed to be published as journal articles in coming years to facilitate the completion of the doctoral dissertations.
In 2017 it is foreseen that three of the scientists who have worked in PANCHO defend their DSc theses: Arkoma, Valtavirta and Myllykylä.
2. Work plan

The work in PANCHO is divided into four work packages: one for each research topics and one for project management.

2.1 Work package 1 (WP1): Computational framework

In SAFIR2014, the FINIX fuel behaviour module \cite{6, 7} has been developed within the PALAMA project. FINIX is a general purpose fuel behaviour module for thermal and mechanical fuel behaviour in multiphysics simulations, and has been integrated into VTT’s Serpent 2 reactor physics code and reactor dynamics codes. In PANCHO, further development and validation of the FINIX fuel behaviour module is a major goal during 2015-2018. Previously FINIX has been verified against FRAPTRAN in certain RIA scenarios and compared against a limited set of experimental fuel temperature data. During the PANCHO project, the development of LOCA capabilities takes the priority. Most significantly this involves extending the current cladding mechanical solution to account for non-elastic deformations so that the plastic deformation occurring in LOCA at high temperature and overpressure can be modelled. This development started in 2015 and continues in 2017. Additions to the steady state capabilities of FINIX were made in 2016. The development continues in 2017 and 2018 in the form of model validation and refinement. Here the goal is to achieve a well validated version of FINIX by the end of 2018 that is applicable to LOCA fuel behaviour simulations as a part of coupled code systems. The validation work is done in collaboration with the FUMAC benchmark and the development of the new validation system.

To better support the FINIX development, the SPACE validation tool will be redesigned. In 2015 a software development plan was written for the new validation tool. This software development plan will guide the programming work and database construction that will be performed in 2016–2017.

A close co-operation is kept between the FINIX development in PANCHO and the proposed SADE and MONSOON projects to implement FINIX to various codes, such as Serpent 2 and TRAB3D. This ensures that the FINIX development focuses on features and issues which are important to end users.

Partners and person months allocated to WP1 in 2015 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>5</td>
</tr>
</tbody>
</table>

2.1.1 Task 1 (T1.1): FINIX

In the first year of PANCHO, the development of the FINIX fuel module focused on improving the user interface and updating the cladding mechanical solution. This made FINIX easier to use also as a stand-alone code, as well as improved the interface of FINIX with other reactor safety codes. The work included updating the FINIX data structures, which improved both the interface and prepares the code for implementation of further models in 2016 and after. In 2016 the steady state solvers were implemented in FINIX. The most important phenomena that are not yet modelled in FINIX include cladding creep, cladding oxidation, cladding ballooning, cladding burst and fission gas release. In 2017 the focus will be in modelling as many of these phenomena as possible and implementing the models in FINIX.

2.1.2 Task 2 (T1.2) Validation system
In 2015 a software development plan was written for the new validation tool. For this purpose, user requirements were gathered, and the purpose, scope, objectives and constraints of the validation code were determined. In 2016 the fuel database was completely redesigned, and programming work of the new validation code was started. In 2017 an array of relevant validation cases will be agreed on and added in the database, and systematic validation of FINIX begins. The user experience from the validation system usage will be analyzed, and the system will be further improved and tested.

2.2 Work package 2 (WP2): Integral fuel behaviour

This work package consists of three tasks: the LOCA and RIA performance of the fuel and the Halden in-kind work.

The successful analysis of the accident behaviour will base on the ability to characterize the changes happening in the fuel during the reactor life. These include the effects of the fission products, the changes of the fuel microstructure, the possible doping and the behaviour of the fission gases. For 2015–2018, goals are developing models for FINIX to support the burnup calculations and creating an in-house fission gas release model.

The goal of the LOCA research in 2015–2018 is to gain a comprehensive and profound picture of nuclear fuel behaviour under LOCA conditions. Of high importance is taking part in international efforts to better model fuel behaviour under LOCA conditions, and to share good principles, data and information on the subject. These activities will support VTT’s in-house model development, verification and validation.

The SCANAIR in-kind work and the participation in proposed OECD/NEA RIA Benchmark Phase III are used to develop and validate the SCANAIR code for use in Finnish reactors. During 2015–2018 the development objectives are attaining the ability to model the BWR materials and conditions as well as implementation of VVER fuel relevant models. The validation and training is concurrent to this development and is done in international context, especially within OECD/NEA RIA fuel codes benchmark.

The fuel-related work of VTT’s in-kind contribution to Halden Reactor Project is managed through task T2.4. This research is funded through the Finland’s agreement with Halden. The research content of the work is agreed annually, and the work is usually reported as Halden Work Reports.

Partners and person months allocated to WP2 in 2015 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>7.5</td>
</tr>
</tbody>
</table>

2.2.1 Task 2 (T2.1): LOCA

Success in LOCA research requires taking part in international efforts. In 2014 an IAEA Coordinated Research Programme “FUel Modelling under Accident Conditions” (FUMAC) was launched, and VTT is participating in this project. Calculations for selected FUMAC cases were performed in 2015 with FRAPTRAN-1.4, to train the experts participating in FUMAC and to serve as a future reference cases for FINIX assessment. In 2017 the FUMAC investigations continue. The last FUMAC research coordination meeting is held in late 2017.

2.2.2 Task 3 (T2.2): RIA

A coupling of an external thermal hydraulics code with SCANAIR was done as an in-kind work for 2015. The VTT in-house general thermal hydraulics code GENFLO was chosen for the coupling. The purpose of this work was to make possible the modelling of BWR thermal hydraulic conditions with SCANAIR, as this
was not possible with the current models in the code. Validation simulations on the coupling were done in 2016. A journal manuscript of the code coupling and validation was prepared, and in 2017 the work performed with RIA and LOCA analysis will be collated to Asko Arkoma’s DSc dissertation.

IRSN has granted VTT the right to use the SCANAIR software on a prerequisite that VTT makes yearly in-kind contribution with the code. The work plan for each year is negotiated in collaboration with IRSN. For 2017, the in-kind contribution will be the participation of the OECD/NEA RIA Benchmark Phase III start.

Participation in the OECD/NEA RIA Benchmark Phase II, begun in 2014 as a part of PALAMA project, was continued in 2015 with the uncertainty analysis of the SCANAIR code. In 2016, the final simulations were performed and one benchmark meeting was participated in. This benchmark phase ended in 2016. The CSNI Activity Proposal Sheet (CAPS) for a new phase of the benchmark will be prepared during the first semester of 2017, and the start is planned for the fall 2017. In the new phase, the same uncertainty and sensitivity analysis methods applied in Phase II are adopted for irradiated fuel. The most important input parameters in irradiated fuel RIA will be distinguished.

2.2.3 Task 4 (T2.3): Halden in-kind

The fuel-related work of VTT's in-kind contribution to Halden Reactor Project is managed through task T2.4. The research content of the work is agreed annually with Halden, and the work is usually reported as Halden Work Reports. The topic for 2017 is tentatively the analysis and reporting of the fission gas release experiment IFA-720.3, which involves two BWR fuel rods, one with standard UO2 and the other with Cr-doped pellets, pre-irradiated Oskarshamn 3 to about 58 MWd/kgU2. The rods are ramp-tested for determination of the fission gas release threshold with the experiment beginning in January 2017 and to be unloaded by mid-2017.

2.3 Work package 3 (WP3): Separate effects

The experimental section of task 3.1 will concentrate on the leaching behaviour of fuel materials in the storage conditions of defective fuel rod. The aim is to produce data for validation of models. The experiments were started in 2015 with the dissolution studies with crystalline ThO2 with similar microstructure to UO2 fuel matrix [12 - 14] which continued the work performed in EURATOM FP7 project REDUPP (Reducing Uncertainty in Performance Prediction).

In 2017 the plan is to conduct leaching experiments using UO2 based SIMFUEL, which contains inactive elements mimicking the behaviour of fission products [11]. SIMFUEL with 50 MWd/kgU burnup contains Sr, Zr, Y, Mo, Ru, Rh, Pd, Ba, La, Ce and Nd. In the matrix of irradiated fuel the fission products can be classified into four groups; oxides dissolved in matrix, metallic precipitates, oxide precipitates and gases (and other volatiles). The elements in the available SIMFUEL samples have members in the first three, groups excluding the group of gases and volatiles. In the experiments the material will be leached in two elevated temperatures and waters mimicking the conditions for storage of defective fuel rod. The release of the different elements into the aqueous phase is followed by sampling and elemental analysis of the solution. The aim is also to gain some information about the surface processes taking place during the leaching. In the leaching experiments conducted with ThO2, an isotopic tracer [15] was used to get insight to the dissolution/precipitation phenomena. The aim is to use some microscopic techniques [9] (SEM, TEM) during the experiments to study the dissolving surfaces [9] and colloids forming in the aqueous phase.

Originally a modelling task was included in task 3.1. However, due to the cuts in the funding this work was left out from PANCHO in 2015 - 2017. In 2017 the development of the chemistry model continues in a parallel project with a half-year grant to Henri Loukusa from the Fortum Foundation. The experimental results gathered in this project can be used to validate the models developed separately for the modelling
of fuel rod chemistry. PANCHO aims to become an associated group to ThermAc project investigating thermochemical properties of nuclear fuel. This will mean right to participate in ThermAc workshops and possibility to present PANCHO research in these meetings.

The task 3.2 builds on the previous modelling work on the creep phenomena, the LCSP model [16-19] and the viscoelastic model [20, 21] that have been developed in past SAFIR projects and in the IDEA project funded by the Academy of Finland. In 2015 the LCSP model was further developed and a doctoral thesis “Modelling nuclear fuel behaviour and cladding viscoelastic response” was prepared and defended. Concurrently planning for an experimental campaign using available materials and experimental facilities was initiated. The experiments currently foreseen are on the stress relaxation and viscoelastic behaviour of zirconium alloys. The experiments are to be conducted in years 2016 – 2017 with on-going modelling effort to both utilize the results and provide feedback. The developed models are also to be implemented into finite element code for 3D analysis of cladding behaviour.

The sources of in-pile data are the experiments performed in Halden reactor and the Melodie test in Osiris reactor as a part of the JHR project. For application of the models to the fuel behaviour codes and the validation of the results, close co-operation with PANCHO project developing fuel code FINIX is maintained.

There are indications that the cladding in- and out-of-pile transient behaviour may differ qualitatively [20]. While the experiments in this project will be conducted in out-of-pile conditions on unirradiated materials, the intent is to have active discussions on possibility of contributing to future in-pile experiments in international projects.

Partners and person months allocated to WP3 in 2015 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>4.5</td>
</tr>
</tbody>
</table>

2.3.1 Task 1 (T3.1): Chemistry

The leaching experiments were started with the analogue material, ThO$_2$, in year 2015. For this material there is already available data from the experiments conducted with crushed ThO$_2$ particles at room temperature [14]. The results obtained in 2015 have been published (in Journal of Radioanalytical and Nuclear Chemistry) and the results were presented as a poster in the International conference on nuclear and radiochemistry held in Helsinki in August 2016. A short memo of the conference was written to further disseminate information presented at the conference.

Plan for 2017 is for Emmi Myllykylä to finish her PhD Thesis work.

2.3.2 Task 2 (T3.2): Cladding mechanical response and load follow

The experiments on the creep behaviour of zirconium alloys were planned to be conducted in years 2016 – 2017 with on-going modelling effort to both utilize the results and provide feedback. In 2016 the LCSP model is further developed to take transient and primary creep into account. The test material (zirconium-base alloy) has been obtained and test specimens have been manufactured for transient creep tests. The results obtained in 2015 and the first half of 2016 have been reported in a journal manuscript and in a conference paper presented in the Baltica X conference in June 2016.

2017 the effects of load follow operations to the fuel are to take the center stage in the investigations. In 2017, the capability to model load follow operations of the fuel behaviour codes used in Finland is reviewed and a state-of-the-art review for creep-fatigue models or models for cyclic mechanical
behaviour will be conducted to explore suitable models for load follow purposes. A fuel failure during a power transient is often caused by an inhomogeneity (e.g. manufacturing flaw) in the fuel pellet causing a localized stress to the cladding. In order to investigate this a 3D model of the fuel is needed. To respond to this need the BISON fuel code from INL is taken into use at VTT, with the first test cases investigating the localized effects to cladding strain distribution.

2.4 Work package 4 (WP4): Management and international co-operation

This work package is used for project management. Along with the management of the substance work in PANCHO, the project plan is updated as per the funding decision, the project results are communicated to the Reference Group and appropriately reported, the project application for the coming years and yearly reports are written.

Partners and person months allocated to WP4 in 2015 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1</td>
</tr>
</tbody>
</table>

2.4.1 Task 1 (T4.1): Project management

Project management tasks of PANCHO are dedicated to this work package. These include preparation of the plans, changing of the plans as per the funding decisions, reporting of the progress and the results of the project, and general management of the project such as project coordination.

This task is also used for the participation in relevant international working groups on fuel behaviour, attending conferences and publication of the results of the research. The current groups are Halden Programme Group (the technical advisory group for Joint Programme of OECD/NEA Halden Reactor Project), OECD/NEA Working Group of Fuel Safety, ETSON Fuel Behaviour Expert Group and FRAPCON and FRAPTRAN Users’ Group. Also the progress of the CABRI project will be followed in PANCHO. The annual participation fee for FRAPCON and FRAPTRAN Users’ Group, $1500 providing the license to use the codes and admittance to the group’s meetings, is paid from this task. Annual review of the international fuel projects is also arranged by PANCHO.

This task will also contain the travel costs of the individual WPs. The travels foreseen are: WGFS and HPG meetings; RIA benchmark Phase III kick-off meeting; IAEA CRP FUMAC research coordination meeting; ETSON fuel EG meeting; potentially fuel behaviour related summer school organized by EC-JRC ITU; BISON workshops organized by INL; TopFuel2017 in Korea; OECD/NEA workshop Paris 7-9 March; ThermAc Workshops; Bulgarian VVER conference in fall 2017.
## 3. Deliverables 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Development of FINIX models; FINIX development update report.</td>
<td>3.5</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D1.2.1</td>
<td>Report on the initial validation of FINIX code using the new system.</td>
<td>1.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>M2.1.1</td>
<td><strong>FUMAC simulations performed</strong></td>
<td>1.5</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Presentation at FUMAC meeting, travel report</td>
<td>0.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D2.2.1</td>
<td>SCANAIR in-kind work for IRSN, proposed items: participation to the proposed RIA benchmark Phase III, report on plan on benchmark participation.</td>
<td>0.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>M2.2.1</td>
<td>Completion of the introduction part for Arkoma’s article based doctoral dissertation</td>
<td>1</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>D2.3.1</td>
<td>Completion of Halden in-kind work, reported in a Halden Work Report series.</td>
<td>4</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>M3.1.1</td>
<td><strong>DSc thesis of Emmi Myllykylä</strong></td>
<td>1</td>
<td>1.10.2017</td>
</tr>
<tr>
<td>D3.2.1</td>
<td>Review (in report format) of fuel performance code capabilities and potential models for load follow operation.</td>
<td>2</td>
<td>1.10.2017</td>
</tr>
<tr>
<td>D3.2.2</td>
<td>Report on the installation and initial investigations with BISON 3D fuel code.</td>
<td>1.5</td>
<td>31.12.2018</td>
</tr>
</tbody>
</table>
D4.1.1 | Travel reports on WGFS and HPG meetings | 0.5 | 31.10.2017
---|---|---|---
D4.1.2 | Reports to SAFIR2018 reference group | 0.5 | Ongoing

**Total pm** | 18

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4. **Project organisation**

The PANCHO project will be performed by VTT Technical Research Centre of Finland Ltd. The project manager is Ville Tulkki (Doctor of Science (in Technology), IPMA-C certified project manager) from VTT. VTT is the responsible organisation for the project.

The main researchers, their organisation, the tasks they will be contributing, and the estimated person months in 2017 are listed in the table below.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Asko Arkoma</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T2.2, T3.2</td>
<td>3</td>
</tr>
<tr>
<td>N.N.</td>
<td>-</td>
<td>VTT</td>
<td>T2.3</td>
<td>4</td>
</tr>
<tr>
<td>Joonas Kättö</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1, T1.2, T2.1</td>
<td>5.5</td>
</tr>
<tr>
<td>Rami Pohja</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T3.2</td>
<td>1.5</td>
</tr>
<tr>
<td>Ville Tulkki</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T4.1</td>
<td>1</td>
</tr>
<tr>
<td>Henri Loukusa</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>2</td>
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<tr>
<td>Emmi Myllykylä</td>
<td>Research scientist</td>
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<td>T3.1</td>
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<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>18</strong></td>
</tr>
</tbody>
</table>
5. Risk management

Like in most research-intensive tasks the main risk in PANCHO is the loss of expertise mid-project in the way of scientists leaving the project. The effect of such an event is attempted to be lessened by a staged approach to project: both initial reviews, plans for experimental campaigns and the experiments themselves as well as modeling solutions should be well documented and documents kept updated.

The other major possible no-go event for individual tasks would be the difficulties obtaining either the samples or the device time for the experiment. This is alleviated both by early surveys on possible sources for the samples and devices and by maintaining a broad initial scope and having alternative solutions.

In case of obtaining material samples a care must be taken in order to ensure the ability of publishing the results. Only publishable work is to be performed.
References


## Expenses

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**Comments:**

Travel expenses are allocated to task T4.1;
Travel to WGFS meeting, HPG meeting, FUMAC and OECD RIA benchmark meetings and conferences (Topfuel2017, EHPG2017, NuMat2017, VVER conference), ETSON Fuel EG meeting; ThermAc Workshops. Also some travel is foreseen in Halden in-kind - this is in the Halden in-kind part.
T4.1 Other includes the costs of computation cluster and software licenses. External services are related to the FRAPCON license.
T3.1 mat&supp and external services are due to the preparation of the chemistry experiment.
SADE
Safety analyses for dynamical events

Hanna Räty, Ville Hovi, Anitta Hämäläinen, Mikko Ilvonen, Ville Sahlberg, Elina Syrjälähti, Veikko Taivassalo
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1. Research theme and motivation

1.1 Background and state-of-the-art

VTT has been at the forefront in the development of the coupled reactor dynamics codes during the last decades, when coupled neutronics-thermal hydraulics codes HEXTRAN and TRAB3D were developed [1,2]. At the moment VTT can perform safety analyses of nuclear reactors with existing in-house programs and such analyses have been continuously performed for safety authorities and power companies. However, the codes’ capability to model new reactor features such as open core geometry is limited. On the other hand, coupling of the CFD to the system codes provides a more detailed analysis of the reactor behaviour.

At the beginning of the last decade VTT coupled the thermal-hydraulics solver GENFLO to NRC’s fuel performance code FRAPTRAN that was a very advanced approach since hydraulics modelling in fuel behaviour codes has traditionally been simple [3,4]. On the other hand, fuel behaviour models used for full core transient analysis have been very crude. As a solution to this problem, VTT developed in the SAFIR2014 program its own lightweight fuel module FINIX, which is specifically designed for modelling of LWR fuel rods in multiphysics simulations. FINIX has been integrated to reactor physics and dynamics codes [5,6] as well as to the VTT-developed CFD-code PORFLO [7].

Multi-physics development is at the moment internationally of great interest in this field. Increased computer capabilities offer possibilities to significant improvements in thermal-hydraulics with CFD level modelling and in neutronics with the Monte Carlo method as in SERPENT [8] code. Full coupled transient modelling with that kind of a body-fitted fine-mesh approach is still in the future, but detailed codes can already now be used for validation and development of models that are used in reactor dynamics codes for practical safety analysis.

Nowadays many thermal-hydraulic system codes are equipped with 3D modules, which can simulate the flow in a component using a coarse 3D nodalization and possibly a porous medium description of the component internals. For example, both ATHLET and RELAP-3D have pseudo 3D modules. The French system code CATHARE has a somewhat advanced 3D module. Typical nodalizations of RPVs in these codes are very coarse, having the order of one thousand nodes. The intention is to resolve large-scale 3D effects, e.g., during core reflooding with a radial power profile. However, spatial discretizations are usually so coarse that the wall functions and the standard turbulence model are not applicable. Accordingly, because of coarse spatial discretizations and limitations of the 3D modules, in many cases correct 3D flow patterns are not obtained and such essential phenomena as mixing are totally ignored. Therefore, to improve the performance of the system codes in 3D flow conditions, many of them have been coupled with CFD codes (e.g., ATHLET-CFX, ATHLET-Fluent, ATHLET-OpenFOAM, RELAP-Fluent).

Furthermore, there are around the world ongoing projects, in which three-dimensional thermal-hydraulics models are coupled with neutronics. VTT has developed its own 3D flow simulation tool PORFLO [7] that is mainly targeted at coupled reactor dynamics applications, in which geometrical complexity does not allow for a CFD-type structure-fitted grid. In the SAFIR2014 PORFLO was used for modelling of the reactor pressure vessel and first simulations together with HEXTRAN/SMABRE were done. Also, a few alternative models for turbulence in a single-phase longitudinal flow in a rod bundle were implemented and tested [8].

To maintain and increase Finnish expertise and safety analysis competence, VTT’s safety analysis codes have to be further developed. Own code development is an excellent way to increase expertise. Additionally, in-house codes permit more freedom in development and coupled use than commercial or open-source codes and without license fees.

1.2 Objectives and expected results

Aim of the project is to model transients and accidents in such a way, that we can give more reliable answers to the safety requirements set in the YVL guides. The main idea is to improve modelling capabilities by validating coupled use of the CFD-type thermal-hydraulics solver PORFLO and the reactor dynamics codes HEXTRAN and
TRAB3D. New submodels for wall friction and mixing are required especially for two-phase conditions. Also the neutronics modelling needs to be more detailed and the whole safety analyses methodology revised to get the full benefit on the improved accuracy of the thermal-hydraulics modelling. The goal is to have a tool, which is more accurate with known uncertainties and still fast and robust enough for practical safety analyses. Own code and in-depth understanding of it enables the best possible expertise on safety analyses.

The developed computational tool set of coupled neutronics, system codes and CFD-type 3D thermal hydraulics will be tested further 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. The objective is to calculate cases in which three-dimensional phenomena are significant, including also cases in which two-phase modelling is required:

- asymmetric flow transients such as pump transients
- asymmetric reactivity transients such as control rod ejection (CRE), control rod withdrawal (CRW)
- sudden changes in coolant conditions such as pressure transients in BWRs, boron dilution, propagation of a cold water front
- Failures in operation or protection such as loss of offsite power (LOOP), load rejection, ATWS
- BWR stability

A further objective is to calculate also transients, which cannot yet be modelled with existing tools

- Mechanical interaction of flow and fuel assemblies: fuel rod bowing, lift off
- Blocked flow channels

At the beginning of the project focus is on VVER plants, but developed methods and tools can be applied also for PWRs and BWRs.

The results of the SADE project for the period 2015-2018 comprise (achievements bolded)

- internally coupled HEXTRAN-SMABRE code, validated for safety analyses
- pin power models in routine use in TRAB3D code
- code sequences Serpent 2-TRAB3D and Serpent 2 - HEXTRAN in routine use
- improved neutronics models in TRAB3D and HEXTRAN
- updating the coupling between the fuel behaviour module FINIX and reactor dynamics codes TRAB3D, HEXTRAN and TRAB-CORE to include the latest version of FINIX, including initialization of fuel rods for the initial state of the transients
- coupled code set of PORFLO and HEXTRAN/TRAB3D-SMABRE in routine use
- coupling between FINIX and PORFLO
- coupling method for two PORFLO simulations with different spatial discretizations
- at least 10 scientific publications (by 2016: 2 scientific publications, 4 conference papers; 2017 planned 2 scientific publications, 2 conference papers)
- 3 master’s thesis and post-graduate degrees (MScTech 2016)
- educated and experienced research staff (especially in coupled analyses and nodal neutronics codes)

The original project plan done in 2014 included also objectives listed below. Due to the remarkably smaller funding, work for these aims cannot be started in 2015-2017.

- models for the influences of wall friction and turbulence in coarse spatial discretizations of open-medium zones; applicable for single-phase and two-phase flows with varying Reynolds number
- porous-medium models for turbulence and its effects in rod bundles for single-phase and two-phase flows with varying Reynolds number
- porous-medium models for the longitudinal and transverse flow resistances in (single-phase and) two-phase flows in rod bundles (varying Reynolds number)
- updated GENFLO-FRAPTRAN code
- own thermal hydraulics in the FINIX module

1.3 Exploitation of the results

This project is part of the work, whose main goal is a fully self-developed safety analyses calculation system that is independent from vendors and codes such as APROS, which are used by power companies. Therefore, STUK can exploit in safety assessments the results that are achieved from the simulations with the tools that are further developed in this project. Also power companies can make use of the knowledge gained during the project. Especially simulations with the coupled 3D thermal-hydraulics and reactor dynamics code system will bring out information that is difficult to get with other modelling tools.
The project consists of several work packages and tools. Some of the results can be applied already during the SAFIR2018 program to safety analyses, e.g. tasks 1.1, 1.2, 2.1 and 3.1, whereas some of the tasks have scientifically more ambitious goals. Such tasks may bring new information of the behaviour of the reactors during transients quite soon, but application to licensing analysis will likely take more time.

1.4 Appropriateness of the project to SAFIR2018 programme

SADE project focuses on modelling of such events in nuclear power plants in which coupled neutronics, core thermal hydraulics and fuel behaviour play an important role. Simulation of these kinds of events is a necessary part of the safety analysis demanded for NPP licencing and requires very nuclear specific expertise and simulation tools. Research content of this project corresponds directly to the demands presented in the sections 3.3.4.4, 3.3.4.1 and 3.3.4.3 of the SAFIR2018 framework plan.

1.5 Education of experts

Research staff in 2017 includes two young researchers, who still require more experience on performing safety analyses. Very limited funding restricts recruiting trainees and younger researchers which would be needed to be prepared for the expected future changes in research staff. Due to the already occurred personnel changes training is urgently needed. However, training of a new expert on nodal codes and neutronics has already begun. First master's thesis in this project has been completed in 2016.
2. Work plan

SADE is organized into four work packages: Neutronics and reactor core (WP1), Open core geometry and RPV mixing (WP2), Hot channel modelling (WP3) and QA and administration (WP4). Work packages 1-3 have both long-running tasks that last for the whole four-year period of the SADE project and shorter tasks. Some tasks will begin later during the project.

2.1 Neutronics and reactor core 1 (WP1)

In this work package neutronics modelling of VTT's three-dimensional reactor dynamics codes is further developed. Development of the solution methods and codes is also an efficient way to study nodal codes in depth. Several topics have been identified that could be further developed.

In 2015 the pin-wise power distribution model in TRAB3D has been improved. Discrepancies in pin-wise and in node-wise power distributions between TRAB3D and Monte Carlo code Serpent 2 were studied [9]. On the base of that work, improvement of axial discontinuity models was started and resulted in a M.Sc. thesis [10] in 2016. In addition, work on the modelling of control rod tips in transient codes has begun in 2016.

Aim is also to have at VTT a fully self-developed calculation system which can be used for the whole calculation sequence from basic nuclear data to coupled 3D transient analyses. The missing piece in VTT's own calculation system has been the preparation of group constants for safety analysis. During the SAFIR2018 the project aim is that group constants can be routinely created with Serpent 2 for transient analysis performed with reactor dynamics codes. In 2015 code sequence Serpent 2 – TRAB3D has been created and stationary simulations have been performed in a HZP state with TRAB3D using group constants created by Serpent 2 and in 2016 the first calculations with Serpent 2 - HEXTRAN code sequence have been performed. In addition to Serpent 2 group constants, the present method for group constant creation with CASMO-SIMULATE is maintained. Achieving these goals demands the training of new experts. This work package will be carried out in close co-operation with the MONSOON project.

Besides providing a way to generate group constants for reactor dynamics analyses, the Serpent 2 code has proved to be a valuable tool in analysing and improving the nodal core models. The accurate reference solutions by Serpent 2 allow to identify submodels where improvement in accuracy in the nodal codes can be achieved.

In this work package also fuel rod modelling in the 3D reactor dynamics codes will be improved. The FINIX fuel module was coupled to HEXTRAN and TRAB3D during the SAFIR2014 program and used for the modelling of PWR and VVER reactor cores. In 2015 work has been continued outside the SAFIR2018 with VTT's own funding. New features implemented to FINIX during SAFIR2018 e.g. in the PANCHO project will be considered in the couplings with reactor dynamics. Changes in the fuel rod dimensions will be taken into account in the flow channel geometry. Further development of the coupling between FINIX and reactor dynamics codes will be continued in late 2017.

Person months allocated to WP1 in 2017 are given in the table below.

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2.1.1 Neutronics model (T1.1)

In this task, the calculation routines in the transient codes TRAB3D and HEXTRAN are analysed, refined and implemented into the codes for immediate use. The current work in this task involves improving the modelling of the control rod tips in TRAB3D and HEXTRAN and the implementation of axially discontinuous fuel model for HEXTRAN. The work on the former has begun and a calculation routine for control rod tips has already been produced. The routine has not yet been attached to TRAB3D or HEXTRAN. In the attaching process the code quality of TRAB3D has been improved and TRAB3D has been made compatible with Windows environment. Axially discontinuous fuel is already implemented in TRAB3D and in order to improve the modelling of more advanced VVER-type reactors such as VVER-1200, HEXTRAN has to be modified to have the same capabilities.

2.1.2 Cross sections (T1.2)

In 2015 and 2016 the work with Serpent 2 - TRAB3D code sequence has focused on stationary calculations in a H2P state with a PWR core. Serpent 2 group constants have been used for VVER-type reactors as well by establishing the Serpent 2 - HEXTRAN code sequence [11]. However, these calculations were in stationary state as well. In 2017 calculations will proceed to full power and time-dependent simulation. The establishment of both Serpent 2 - TRAB3D and Serpent 2 - HEXTRAN code sequences enable the choosing of possible transient cases from PWR, BWR and VVER cases. This work will be carried out in close co-operation with the MONSOON project.

Serpent 2 and HEXTRAN code sequence will be presented at the M&C 2017 conference (Mathematics and Computational Methods Applied to Nuclear Science and Engineering). In 2017 the progress of this task is dependent on the available resources in the MONSOON project.

2.2 Open core geometry and mixing in RPV (WP2)

This work package focuses on the whole core transient analyses especially in cases where mixing in a reactor pressure vessel and open core geometry play an essential role. Computational tools that enable more realistic modelling of the transients will be further developed and validated simulating transients.

The primary solution to the analysis needs that arise from open core geometry e.g. in AES-2006 plant is the internally coupled HEXTRAN-SMABRE. The original coupling between the reactor dynamics code HEXTRAN and the system code SMABRE is designed for cases in which flow in a reactor core propagates in separate flow channels. In that parallel coupling both codes calculate thermal hydraulics in the core. In HEXTRAN each fuel assembly is usually modelled with individual flow channels whereas SMABRE models core with fewer channels. Internal coupling between TRAB3D and SMABRE was implemented during the predecessors of SAFIR2018, but that coupling could be used only in modelling of BWRs and PWRs with quadratic fuel assemblies [12]. In the internal coupling, SMABRE takes care of the hydraulics calculation in the whole cooling circuit including the reactor core. The first version of the internally coupled HEXTRAN-SMABRE was done in 2015 and the coupled code has been applied successfully to the first test simulations of VVER-440 and VVER-1000 transients.

HEXTRAN-SMABRE model for the Kalinin-3 VVER-1000 plant will be created later during the project. The model will be used for testing of the internally coupled HEXTRAN-SMABRE. It will be used also in the SAFIR2018/USVA project, in which it is needed for the modelling of the UAM benchmark exercise.

More advanced but at same time challenging modelling approach was the coupling of reactor dynamics codes with fully three-dimensional thermal-hydraulics. The preliminary one-way coupling between HEXTRAN-SMABRE and PORFLO was done in the KOURA project of SAFIR2014. In 2015 the work continued in the SADE project with the two-way coupling between HEXTRAN-SMABRE and PORFLO. The two-way coupled HEXTRAN-PORFLO-SMABRE code system was completed in 2016. The communications between codes are visualized in Figure 1. A CFD mesh covers a RPV or part of it and a SMABRE model the rest of the plant. HEXTRAN communicates with PORFLO but not with SMABRE. Two benchmark cases, a VVER-1000 MSLB transient (OECD/NEA benchmark V1000CT-2) and the 7th dynamic AER benchmark for VVER-440, were computed successfully with the new coupled HEXTRAN/SMABRE and PORFLO code system. Already the first coupled simulations provided more realistic results and demonstrated the feasibility and advantages of coupling real 3D thermal hydraulics with a system code [13].
In 2017-2018, the development of the two-way coupled HEXTRAN-PORFLO-SMABRE computation system will continue. The validity of the new coupled modelling framework in whole-plant safety analyses will be examined. The coupling system will be improved, streamlined and completed for different kind of coupled analyses to be performed. The simulations with the coupled codes will be published in papers, conferences and AER symposia.

In 2017, two transient computations will be performed with the two-way coupled HEXTRAN-PORFLO-SMABRE code system. A new mesh covering the whole RPV will also be created for VVER-440 in 2017. In addition, the VVER-1000 meshes will be extended to cover also the reactor head.

In order to speed up computations, an OpenFOAM based solver will be utilized in coupled simulations in 2018. In addition, to model thermal hydraulics in the multiphase cases, an advanced version of the 5-equation approach ("multiphase mixture model") will also be employed in 2018.

Modelling of plant transients with a detailed CFD-type body-fitted mesh is in practice unattainable and thus a coarse mesh together with the porous medium representation is needed. Use of the coarse mesh requires scrutinization of closure models e.g. for wall friction and especially for mixing phenomena. The SADE project will focus on couplings between CFD, neutronics and system codes and simulations with the coupled code system. The proposed but rejected MICSA project was planned to concentrate on improving the modelling of coolant mixing in porous-zone and coarse-mesh representations of perforated plates and tube bundles.

Person months allocated to WP2 in 2017 are given in the table below.

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2.2.1 Internally coupled HEXTRAN/SMABRE (T2.1)

Further development and validation of the internally coupled HEXTRAN-SMABRE is on hiatus in 2016-2017.

2.2.2 3D thermal hydraulics modelling (T2.2)

In 2017 application and validation of the 2-way coupled HEXTRAN-PORFLO-SMABRE reactor analysis framework will be continued. Modelling of 3D thermal hydraulics in a RPV or in a part of it with the CFD-code PORFLO will be performed as coupled to the SMABRE model representing the rest of the plant and on the other
hand coupled to HEXTRAN calculating neutronics (cf. Figure 1). The PORFLO computations will be carried out as any CFD modelling but applying a turbulence model with a porous-medium correction [7].

The objective of this task is to validate the 2-way coupled HEXTRAN–PORFLO-SMABRE simulation framework in safety analyses. Furthermore, simulations with the coupled CFD-system code system will clearly provide more realistic results for several asymmetric VVER-440 and VVER-1000 transients studied in the international benchmarks. The results will be internationally novel and will be reported in papers and in the AER symposium.

The following whole-plant simulations will be performed applying the 2-way coupled HEXTRAN-PORFLO-SMABRE analysis framework:

1. 7th AER benchmark (a VVER-440 loop reconnection): re-computation using a new coarse mesh (whole RPV)

2. OECD/NEA dynamic benchmark V1000CT-2 - scenario 2 (asymmetric VVER-1000 MSLB transient): coarse mesh

In addition, since the existing VVER-440 mesh does not cover the core and the upper plenum of RPV, a new computational mesh for the whole RPV will be created. The VVER-1000 meshes will be completed to cover the reactor head.

Simulation results will be compared to the HEXTRAN/SMABRE results and to results reported in literature.

2.3 Hot channel modelling (WP3)

Important outcome of the safety analyses is information about possible fuel rod failures and margins to the failure. Thus in addition to the whole core transient calculations, most demanding circumstances during transients have to be analysed in more detail. DNB (departure from nucleate boiling), CPR (critical power ratio) and fuel rod failures are normally analysed with separate calculations without coupling to neutronics. In work package 2 methods for these analyses will be developed.

Aim of this work package is to improve modelling by extending analysis from traditional separate hot channel and hot rod analyses. At the beginning of the last decade VTT has coupled NRC’s fuel rod transient code FRAPTRAN to the thermal hydraulics code GENFLO [4] and since then FRAPTRAN-GENFLO has been used for safety analyses. In the PALAMA project of the SAFIR2014 program this coupling was renewed so that GENFLO-FRAPTRAN can now model several fuel rods [14]. Modelling with the FRAPTRAN-GENFLO and GENFLO-FRAPTRAN has revealed some limitations in the programs and thus some properties of the code demand modifications. For example FRAPTRAN-GENFLO analyses have revealed that some challenging situations e.g. in LOCA cases could be simulated more reliably by improving the treatment of boundary conditions originating from a plant simulation. Interaction of several fuel rods will be analysed in this project, e.g. in the case of blockage in fuel assembly. FINIX is VTT’s fuel behaviour module that is designed for modelling of LWR fuel rods in multiphysics simulations. The FINIX code will be supplemented with a thermal hydraulics model, which allows stand-alone use of FINIX.

Also the effect of mixing and cross-flows in fuel assemblies and between assemblies will be analysed with the 3D hydraulics solver. The whole DNB/CPR methodology will be revised so that it can be applied to porous media CFD-simulations. SADIE project will work in co-operation with the NURESA project, especially during latter half of the SAFIR2018 program.

Aim is that in the future instead of the FRAPTRAN-GENFLO coupling, VTT’s own fuel performance module FINIX could model the fuel behaviour, and flow conditions could be analysed with a real 3D code, such as PORFLO. One step towards this goal is supplementing of FINIX with its own hydraulics model. An integrated hydraulics model enables stand-alone validation of FINIX and provides more possibilities for modelling in the coupling with such codes, which do not have their own thermal hydraulics models.

3D thermal-hydraulics in a hot channel and its vicinity will be studied by means of porous-media CFD modelling. Computations will be carried out for an open-core channel starting with the single-phase or low-void flows. Later on conditions with an increasing void fraction will be studied. Boundary conditions will be obtained from whole-core simulations. In CFD, when more detailed modelling is desired in a certain area, the most usual method is a refinement of the mesh in that area. However, mesh generation in general is not straightforward, and the inclusion of a refined area may sometimes bring new problems, if not done properly. Another way to ‘zoom in’ on an interesting area (like hot fuel assembly) is to couple two CFD simulations, with the larger domain ‘feeding’ the smaller one with boundary conditions. In this project work is started with this latter method.

This work package was not started in 2015-2016 due to the inadequate funding and can not be started in 2017 either.
2.3.1 Hybrid mesh CFD modelling (T3.2)

Not done in 2017.

2.4 QA, Co-operation and administration (WP4)

This work package consists of tasks, which support actual research tasks in work packages 1-3. Work Package 4 contains project management and administration and international activities tasks. The latter include participation in conferences, training courses and relevant benchmarks. The work package contains also task that is intended for the maintenance and documentation of VTT’s reactor dynamical calculations system.

Person months allocated to WP4 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>0.9</td>
</tr>
</tbody>
</table>

2.4.1 International co-operation (T4.1)

Participation in the OECD Nuclear Energy Agency (NEA) working groups and benchmarks has been one of the most important ways of validating the methods and codes used in reactor analysis. This task has included participation in the activities of the NEA Working Party on the Scientific Issues of Reactor Systems (WPRS), which is responsible for the organization of the reactor dynamics benchmarks among other activities. In 2017 WPRS meetings will not be attended.

The cooperation and information exchange on VVER safety within the AER framework will be continued. AER is an association for research institutes and power companies working with VVER reactors. Nominal membership fee guarantees free access for all VTT’s researchers to AER’s activities, including its annual symposium. Significant activities are also groups on different topics. For the SADE project the most interesting is the working group D on safety analyses, which e.g. has organized several reactor dynamical benchmarks. In addition, a relatively new working group G on nuclear applications of three-dimensional thermal hydraulics relates to activities of the SADE project.

2.4.2 QA and documentation (T4.2)

The task aims at reporting the research results of the project, improving the usability of the code system e.g. through proper documentation and pre- and postprocessors, as well as making it possible to perform some other necessary limited development work that cannot be foreseen. The results of the work done during the SAFIR2018 program and its predecessors are published as VTT research reports as well as in international journals and conferences. In 2016-2017 this task is on hiatus.

2.4.3 Project administration (T4.3)

The task consists of project administration work, reporting duties for the SAFIR2018 programme, including in 2017 also the mid-term publication and seminar, reporting to the reference groups and attending the project meetings as well as detailed planning of the SADE project for the year 2018.
### 3. Deliverables and milestones 2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Short description of the deliverable or milestone.</th>
<th>Criterion for the approval of the milestone.</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Successful computation with TRAB3D or HEXTRAN with the new control rod tip model</td>
<td></td>
<td>Postponed in favor of implementing axially discontinuous fuel model to HEXTRAN code</td>
<td>0.75</td>
<td>N/A</td>
</tr>
<tr>
<td>D1.1.2</td>
<td>Conference paper or report on an example calculation with the new control rod tip model</td>
<td></td>
<td>Postponed in favor of implementing axially discontinuous fuel model to HEXTRAN code</td>
<td>0.25</td>
<td>N/A</td>
</tr>
<tr>
<td>D1.1.3</td>
<td>Successful porting of HEXTRAN to Windows/Visual Studio environment for developmental purposes</td>
<td></td>
<td>Approval criterion: a successfully working code version</td>
<td>0.25</td>
<td>30.6.2017</td>
</tr>
<tr>
<td>D1.1.4</td>
<td>Successful calculation with HEXTRAN with axially discontinuous fuel model.</td>
<td></td>
<td>Approval criterion: SAFIR report on the successful calculation, approval of SAFIR reviewer</td>
<td>2.3</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D1.2.1</td>
<td>Comparison of group constants from CASMO-4 and Serpent 2 for transient calculations</td>
<td></td>
<td>Postponed in favor of implementing axially discontinuous fuel model to HEXTRAN code</td>
<td>0.5</td>
<td>N/A</td>
</tr>
</tbody>
</table>
| D1.2.2 | Transient calculation using group constants generated with Serpent 2, in co-operation with MONSOON project.  
Postponed in favor of implementing axially discontinuous fuel model to HEXTRAN code | 0.7 | N/A |
| D1.2.3 | Presentation of Serpent 2 and HEXTRAN code sequence at M&C 2017 conference  
Approval criterion: Conference full paper review, approval of SAFIR reviewer | 0.25 | 30.4.2017 |
| D1.2.4 | Paper on Serpent-HEXTRAN code sequence in KERntechnik  
Approval criterion: Scientific journal’s full paper review, approval of SAFIR reviewer | 0.1 | 1.9.2017 |
| D2.2.1 | New computational meshes for VVER-440 covering the whole RPV  
The existing VVER-440 meshes do no cover the core and upper plenum. A new set of meshes will be creating for the whole VVER-440 RPV.  
Approval criterion: presented to the reference group | 0.5 | 30.4.2017 |
| D2.2.2 | Completion computational meshes for VVER-1000 to cover the whole RPV (NEW)  
The existing VVER-1000 meshes do no cover the reactor head. A new set of meshes will be creating for the whole VVER-1000 RPV.  
Approval criterion: presented to the reference group | 0.3 | 30.4.2017 |
| D2.2.3 | The coupled HEXTRAN-PORFLO-SMABRE simulations will be presented in the AER working group D meeting | 0.15 | 11.5.2017 |
| D2.2.4 | Coupled re-computation of AER-BM7 with the new coarse whole-PRV mesh  
Coupled HEXTRAN-PORFLO-SMABRE simulation of AER-BM7 will be performed using the new coarse VVER-440 mesh  
Approval criterion: presented to the reference group | 0.35 | 30.6.2017 |
<table>
<thead>
<tr>
<th>Task Description</th>
<th>Effort</th>
<th>Date</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coupled computation of AER-BM6 with the coarse mesh</td>
<td>0.35</td>
<td>30.8.2017</td>
</tr>
<tr>
<td>Coupled computation of the second scenario of the asymmetric VVER-1000 transient benchmark with the coarse mesh</td>
<td>0.3</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>The coupled HEXTRAN-PORFLO-SMABRE simulations will be presented in the AER symposium</td>
<td>0.45</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>Coupled computation of the second scenario of the asymmetric VVER-1000 transient benchmark with the less coarse mesh</td>
<td>0.2</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>D2.2.5</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Coupled computation of the first scenario of the asymmetric VVER-1000 transient benchmark with the coarse mesh</td>
<td>0.35</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>Coupled HEXTRAN-PORFLO-SMABRE simulation of Scenario 2 of Exercise 3 in the OECD/NEA MSLB benchmark V1000CT-2 will be performed using the coarse mesh</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Approval criterion: presented to the reference group</td>
<td></td>
<td></td>
</tr>
<tr>
<td>D2.2.6</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Paper on the coupled HEXTRAN-PORFLO-SMABRE simulations for an asymmetric VVER-1000 transient (coarse mesh)</td>
<td>0.7</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>Coupled computation of AER-BM7 with the less coarse mesh</td>
<td>0.2</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>Coupled computation of the first scenario of the asymmetric VVER-1000 transient benchmark with the less coarse mesh</td>
<td>0.25</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>D4.1.1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Travel reports etc.</td>
<td>0.3</td>
<td>31.1.2018</td>
</tr>
<tr>
<td>Total pm</td>
<td></td>
<td>6.6</td>
</tr>
</tbody>
</table>

D4.3.1 Research plans, progress reports, annual reports, SAFIR2018 mid-term seminar etc. | 0.6 | 31.1.2018 |
4. Project organisation

The SADE project will be carried out at VTT Technical Research Centre of Finland. MSc Hanna Räty will act as a project manager until 8/2017 due to MSc Elina Syrjälähti’s parental leave. The research trainee working in the project graduated in 2016, and later it would be necessary to recruit more trainees, depending on project funding. The project members, their participation by tasks and the estimated person months for 2017 are summarized in the table below.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ville Hovi</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T2.2</td>
<td>0.4</td>
</tr>
<tr>
<td>Anitta Hämäläinen</td>
<td>Research team leader</td>
<td>VTT</td>
<td>T2.2, T4.1</td>
<td>0.3</td>
</tr>
<tr>
<td>Mikko Ilvonen</td>
<td>Principal scientist</td>
<td>VTT</td>
<td>T2.2</td>
<td>0.2</td>
</tr>
<tr>
<td>Hanna Räty</td>
<td>Project manager until 8/17, Senior Scientist</td>
<td>VTT</td>
<td>T1.2, T2.2, T4.1, T4.3</td>
<td>0.6 (on sick leave -&gt; 4/17)</td>
</tr>
<tr>
<td>Ville Sahlberg</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1, T1.2, T4.1, T4.3</td>
<td>3.1</td>
</tr>
<tr>
<td>Elina Syrjälähti</td>
<td>Project Manager from 8/17, Senior scientist</td>
<td>VTT</td>
<td>T1.2, T2.2, T4.3</td>
<td>0.3 (on parental leave -&gt; 8/17)</td>
</tr>
<tr>
<td>Veikko Taivassalo</td>
<td>Principal scientist</td>
<td>VTT</td>
<td>T2.2</td>
<td>1.7</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>6.6</strong></td>
</tr>
</tbody>
</table>
5. Risk management

Beyond the applied VYR funding and the contribution of VTT, SADE does not rely on any external funding. Therefore, there is no excess risk related to project funding.

For personnel resources, the current state of the nuclear energy field in Finland may imply a risk of increased competition for experienced personnel. Competition of the personnel has already led to the present situation, in which education is needed especially in nodal neutronics codes. Also absences as long sick leaves may complicate reaching the research goals.

Some research topics of the SADE project demand very computation-intensive modelling. When simulations take a long calendar-time, possibly emerging difficulties in the simulations and with computers may lead to delays in the whole project. For that reason coupled 3D phenomena are in the SADE project modelled at the same time with several distinct approaches to get both short-term and long-term solutions to the analysis needs.
References


7. Ilvonen M., Hovi V. and Taivassalo V., 3D Core thermal hydraulics with the PORFLO code – turbulence modelling and porous medium with porosity steps, in proceedings of 22nd International Conference on Nuclear Engineering ICONE-22, Prague, Czech Republic, July 7-11, 2014.


11. Sahlberg, V., Recalculating the steady state conditions of the V1000 zero power facility at Kurchatov Institute using Monte Carlo and nodal diffusion. To be published in the Proceedings of the 26th symposium of AER, Helsinki, Finland, 10-14 October 2016, MTA Centre for Energy Research.


13. Hovi, V., Taivassalo, V., Hämäläinen, A., Syrjälähti, 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. To be published in the Proceedings of the 26th symposium of AER, Helsinki, Finland, 10-14 October 2016, MTA Centre for Energy Research.

## SADE
Safety analyses for dynamical events

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel Mat&amp;supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Memb fee</th>
<th>Other</th>
<th>TOTAL</th>
<th>VYR</th>
<th>VTT</th>
<th>Other</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>person mo</td>
<td>keuro</td>
<td>keuro</td>
<td>keuro</td>
<td>keuro</td>
<td>keuro</td>
<td>keuro</td>
<td>keuro</td>
<td>keuro</td>
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</tr>
<tr>
<td><strong>WP1 - Neutronics and reactor core</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>T1.1 Neutronics model</td>
<td>2.86</td>
<td>26.8</td>
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<td>29.4</td>
<td>20.5</td>
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<tr>
<td>T1.2 Cross sections</td>
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<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
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<tr>
<td><strong>WP2 - Open core geometry and mixing in RPV</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T2.1 Internally coupled HETTRANSMABRE</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0</td>
<td>0</td>
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<tr>
<td>T2.2 3D thermal hydraulics modelling</td>
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<td>39.04</td>
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<td>0</td>
<td>41.6</td>
<td>29.0463</td>
<td>12.5137</td>
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<tr>
<td><strong>WP3 - Hot channel modelling</strong></td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>T3.1 Hybrid mesh CFD modelling</td>
<td>0.0</td>
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<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
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<tr>
<td><strong>WP4 - QA, Co-operation and administration</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T4.1 International co-operation</td>
<td>0.93</td>
<td>12</td>
<td>1</td>
<td>8</td>
<td>0</td>
<td>0</td>
<td>1</td>
<td>22</td>
<td>15</td>
<td>7</td>
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<tr>
<td>T4.2 QA and documentation</td>
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<td>4.1</td>
<td>0.3</td>
<td>7.5</td>
<td>0.3</td>
<td>0</td>
<td>12.2</td>
<td>8.52656</td>
<td>3.67342</td>
<td></td>
</tr>
<tr>
<td>T4.3 Project administration</td>
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<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td>6.6</td>
<td>78.1</td>
<td>6.0</td>
<td>7.5</td>
<td>0.0</td>
<td>0.3</td>
<td>93.0</td>
<td>65.0</td>
<td>28.0</td>
<td>0.0</td>
</tr>
</tbody>
</table>

Comments:
Publication costs are allocated to the respective tasks.

Mat & supp:
VTT research facility costs (computer clusters), 5.8 € / h of project person time.

Memb fee: AER membership fee 300 € /y.
SAFIR2018 Project plan

USVA
Uncertainty and sensitivity analyses for reactor safety

Torsti Alku, Asko Arkoma, Elina Syrjälähti, Ville Valtavirta

VTT Technical Research Centre of Finland
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   1.3 Exploitation of the results .........................................................4
   1.4 Appropriateness of the project to SAFIR2018 programme ..............4
   1.5 Education of experts ...............................................................4

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      2.1.2 Methodology for determining input uncertainties (T1.2) ..........7
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1. Research theme and motivation

During the SAFIR2018 programme, the uncertainty and sensitivity analyses for reactor safety are proposed to be performed collectively in the USVA project. 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. USVA promotes activities at the interfaces of the different disciplines in reactor safety. The goal of the project is to develop sensitivity and uncertainty analysis methods and practices for multiphysics problems in reactor safety and to train experts in this area.

1.1 Background and state-of-the-art

Traditionally deterministic safety analyses in nuclear safety have relied on conservative models and parameter values. Here the model or input value in relation to acceptance criteria is deliberately chosen conservatively to take into account all the uncertainty affecting the modelling of the system. Until the latest renewal of the Finnish Regulatory Guides on nuclear safety (YVL Guides, 2013), the regulations required the use of conservative assumptions in safety analyses.

The recently updated YVL guides grant the possibility to use the best estimate plus uncertainty (BEPU) methodology instead of conservative analyses. In many cases this is advantageous. On the other hand, the methods for performing such analyses, especially in the cases of coupled simulation codes and long calculation sequences, still require development. Based on the existing international work, it is clear that more investigation into uncertainty propagation methods is needed. This is true for both coupled code systems and calculation chains consisting of successive simulations. Meanwhile challenges still remain, for example, in determining the distributions of the input parameters of a model and in analysing complex computer codes with large uncertainties. Safety analyses should also be accompanied by sensitivity analyses to identify the most important factors in the uncertainties of the calculation results.

Internationally the studies in uncertainty analysis within nuclear safety are often carried out in the context of benchmark programmes such as OECD/NEA Best Estimate Methods Uncertainty and Sensitivity Evaluation (BEMUSE) and OECD/NEA Benchmarks for uncertainty analysis in modelling (UAM) for the design, operation and safety analysis of LWRs (Ivanov et al. 2013). The members of USVA have had an active participation in these communities, for example by attending the UAM-LWR benchmark meetings since the first UAM workshop in 2007. The previous work has yielded results of a high international level in several areas of uncertainty and sensitivity analysis (Pusa 2012a, 2012b, 2013, 2014, Ikonen and Tulkki 2014, Vanhanen 2014a, 2014b, 2015).

To go beyond the state-of-the-art, USVA not only continues the on-going work, but also works towards finding methods for multiphysics and calculation sequence analyses. This is a recently emerged subject that is at the moment extremely topical and fertile for new developments. The work on this topic includes developing methods and practices for determining input uncertainties, propagating uncertainty and finding priorities for uncertainty reduction in multiphysics calculations through sensitivity analysis of coupled codes. These are both topics to be explored in USVA.

In 2015, USVA had a high scientific level with a total of five peer-reviewed scientific papers (Ikonen 2015b, Ikonen 2015c, Arkoma and Ikonen 2016a, Pusa 2016, Vanhanen and Pusa 2015d). In addition, one special assignment was completed (Taavitsainen 2015). The most important accomplishments of the project in 2015 included a thorough comparison on statistical sensitivity analysis methods in the context of fuel behavior modelling (Ikonen 2015b, Ikonen 2015c), sensitivity analysis of local uncertainties in a 5-code calculation chain of a large break loss of coolant accident (LB-LOCA) (Arkoma and Ikonen 2016a) and the development of an automated uncertainty analysis calculation system for the CASMO-4 – SIMULATE-3 calculation chain (Pusa 2016). In 2016, two scientific papers were prepared: conference paper on statistical and sensitivity analysis on LOCA (Arkoma and Ikonen 2016b), and a journal manuscript on CASMO-4 – SIMULATE-3 uncertainty propagation sequence (Pusa and Isotalo 2016). Also, a Master’s thesis was finalized at Aalto University (Taavitsainen 2016). A more detailed description of the research carried out in 2016 is given in Section 2.
1.2 Objectives and expected results

The general goal of the USVA project is to develop methods and practices in uncertainty and sensitivity analyses of multiphysics problems and calculation sequences in reactor safety. The goal supports the long-term aim of establishing a comprehensive methodology for uncertainty and sensitivity analysis for the entire reactor safety field.

USVA approaches the general goal on several fronts. The problem of performing a complete analysis of the entire coupled code system involving uncertainties from various stages of the calculation sequence and from the different physical sources is simply too complex. For this reason, the problem is analysed in parts. These parts form the concrete objectives of the project.

During the four-year period, USVA is expected to advance the knowledge in the analysis of multiphysics coupled calculations by studying both steady state and transient behavior of the reactor. Methods for propagating uncertainties through calculation sequences will be developed, with applications in reactor dynamics calculations and in fuel rod failure analyses. The methods will be implemented in safety analysis codes and tools at VTT. These objectives are also supported by the knowledge gained from individual code and accident scenario analyses, and by the methods developed for input uncertainty estimation. In addition, USVA aims at developing an effective practice for uncertainty and sensitivity analyses of multiphysics systems, such as combined neutronics and thermal hydraulics or fuel behavior simulations. In addition, the most influential uncertainty groups will be identified in such simulations.

1.3 Exploitation of the results

Utilizing the best estimate plus uncertainty (BEPU) approach in reactor safety analyses requires knowledge on the uncertainties of the input parameters and methods to propagate those uncertainties through the simulation models and calculation sequences. In both areas, there are unresolved problems that USVA aims to solve by providing tools for the estimation and propagation of input uncertainties.

The recently updated YVL Guides enable the use of BEPU methods in licensing and safety analyses. However, in all applications involving more than a single analysis code, the methods and practices are still in need of development. The development of practical but rigorous licensing tools cannot take place without a thorough understanding of the models and their properties. For this reason, the results of the USVA project should be highly interesting to both STUK and the utilities.

By the end of the project, it is expected that the developed methods will be in use as safety analysis tools. In addition, the knowledge gained from the analyses of multiphysics simulations and calculation sequences can be used in estimating the impact of the uncertainties of an individual model or a code.

1.4 Appropriateness of the project to SAFIR2018 programme

USVA is a project that is at the heart of reactor safety, operating in the areas of reactor physics, fuel and thermal hydraulics. It also aims at improving the tools and expertise in BEPU safety analyses, which is a very important and timely topic in Finnish reactor safety research.

The need for developing tools and practices for uncertainty and sensitivity analysis is explicitly mentioned in the SAFIR2018 framework plan (Annex 1). Here it is emphasized that to be able to analyse results from coupled or interdependent codes using coherent methods and practices is increasingly important. This is exactly the objective that USVA is striving to achieve.

1.5 Education of experts

The increasing need for experts in nuclear power plant safety and in uncertainty analysis in particular is recognized in USVA. The project concept itself facilitates learning across traditional boundaries, both between disciplines and institutions. Cross-disciplinary learning will be reinforced by project meetings and seminars. In addition, the project also takes a very hands-on approach to education of experts. In the first year, one special assignment was completed and a master's thesis was started. The master's thesis was completed in the beginning of 2016. Furthermore, several project members are working towards their doctoral theses and USVA will have an integral role in those efforts. In 2017, one member of the USVA project team (Asko Arkoma) is foreseen to defend his doctoral thesis.
2. Work plan

USVA is organized into three Work Packages (WPs): Methods and analyses (WP1), Multiphysics and the calculation chain (WP2), and Administration (WP3). The research and analysis work is carried out in the first two WPs, and project administration in WP3. Each of the Work Packages 1-2 has at least one long-running task that lasts the whole four year period of USVA. This enables the project to tackle demanding long-term objectives and provides continuity over the duration of the project. In addition, the work packages contain one-off tasks that can be completed within a shorter period (such as Master’s thesis projects), or tasks that recur after sufficient progress has been made in parallel Work Packages.

In WP1, methods and tools for sensitivity analysis are developed. The work includes both the development of theoretical analysis methods and the implementation of them in computational tools and codes. The tools are also utilized in the analysis of simulation and application data in WP1 and later in WP2. The knowledge obtained in the different disciplines in WP1 is brought together in WP2. In WP2, the project applies the methods and tools in multiphysics and coupled code analyses. This involves both applications using concurrent coupling, such as multiphysics simulations of coupled reactor physics and fuel behaviour, and the analysis of propagating the uncertainties in a hierarchical coupling, such as the reactor simulation calculation chains. The analysis of coupled codes requires good understanding on the individual codes and knowledge of suitable methods, and thus the WP2 is going to interact closely with WP1. Over the course of the four-year project, the tasks gradually move towards more challenging applications. Methods for propagating uncertainties between codes are developed and knowledge on the individual parts is built up. This facilitates the analyses of multiphysics systems and the calculation chains, and the focus of the project continuously shifts towards Work Package 2.

The planned funding for USVA is shown in Annex 2-1 for year 2017 and in Annex 3 for 2015-2018.

2.1 Methods and analyses 1 (WP1)

Work Package 1, Methods and analyses, consists of tasks where methods and tools for uncertainty and sensitivity analysis are developed, and where these tools are applied to real-world data. For the four year duration of USVA, there are two continuously running tasks, one beginning in year 2015, and another in 2016.

In the task T1.1, the existing data on APROS and FRAPTRAN simulations of an EPR loss of coolant accident is analysed. A previous statistical analysis on the rod failures (Arkoma et al. 2015) was extended in 2015 by performing sensitivity analyses to determine the underlying cause of rod failures (Arkoma and Ikonen 2016a). In particular, the effect and importance of various local parameters, i.e. the location related parameters and the sampled fuel manufacturing parameters, to the outcome of chosen output parameters was studied. The results were further disseminated in TopFuel 2016 conference (Arkoma and Ikonen 2016b), including some updates to the previous results (Arkoma et al. 2015, Arkoma and Ikonen 2016a). In 2016, the previously done neural network analysis of FRAPTRAN fuel behavior simulations (Arkoma et al. 2015) is extended with support vector machines (SVMs). This cures the problem of poor statistics which prevented a full analysis with the neural network model. In addition, an SVM surrogate model of the FRAPTRAN simulations is developed which can be used in sensitivity analysis and rod failure prediction. Finally, in years 2017-2018, the global boundary conditions calculated by APROS will be included in the analysis. With the developed SVM machinery, it is possible to analyse the combined effect of the APROS and FRAPTRAN simulations on the rod failure probability. The investigation therefore naturally moves towards multiphysics and WP2 in the last two years of the project.

The second continuous task started in the year 2016. The aim is in developing a generic methodology for determining input uncertainties in nuclear safety codes. In coupled calculations, the uncertainties related to the modelling of thermal-hydraulics arguably dominate the overall uncertainty in the calculation results, for which reason their estimation should have the highest priority in uncertainty analysis. However, comprehensive methodologies for determining these uncertainties have not been accomplished yet. Therefore, there is a clear need for
further research in this area. The methodology to be developed in USVA would use experimental data and code calculations to evaluate the uncertainties in code models. In so doing the task would continue in the footsteps of the work performed within the SAFIR2014 UBEA project, but improve and generalize the area of usability of the results. The UBEA project included the participation in the OECD/NEA PREMIUM benchmark, which was an international effort into determining applicability of evaluating the uncertainties in thermal hydraulic codes’ closure equations in reflooding phenomena with current methods. Ultimately the development of a new methodology would provide a standard tool that could be used with different kinds of nuclear safety analysis tools and could potentially be a solution for the issues ran into in the PREMIUM benchmark. A direct use case for the methodology would be for example its use in conjunction with Apros to facilitate verification and validation of the code within the SAFIR2018 project COVA.

In addition to the continuous work, WP1 was also planned to support several shorter tasks in the years 2015-2018. Many of these are linked to the on-going OECD/NEA UAM-LWR benchmark. In 2015, one such task continued the previous sensitivity analyses on fuel behavior modeling (Ikonen and Tulkki, 2014). In the task, statistical sensitivity analysis methods were compared. The work was presented in the TopFuel 2015 conference (Ikonen 2015b) and submitted to Nuclear Engineering and Design (Ikonen 2015c).

In 2017 the sensitivity/uncertainty work based on generalized perturbation theory (GPT) will be restarted through the implementation of a collision history based GPT technique to Serpent 2. The methodology, derived by Manuele Aufiero based on an extended version of Serpent 2.1.19 (released in March 2014) has quickly gained a foothold as a regular sensitivity calculation tool (see, e.g. Kodeli 2016) and will be finally implemented to the official Serpent release in WP1. The methodology has applications in standard nuclear data sensitivity/uncertainty (s/u) calculations as well as in coupled multi-physics s/u analysis and the analysis of uncertainty propagation in Monte Carlo burnup calculations. The newly implemented methodology will be demonstrated and verified with sensitivity calculations in the framework of the OECD/NEA UAM-LWR benchmark. The work started in 2017 is expected to be continued also in 2018.

Partners and person months allocated to WP1 in 2017 are given in Table 1 below.

Table 1 Partners and person months in WP1

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>7.5</td>
</tr>
</tbody>
</table>

2.1.1 Analysis of rod failures in LB-LOCA (T1.1)

A large break loss of coolant accident (LB-LOCA) in an EPR type power plant has been previously evaluated at VTT with statistical methods (Arkoma et al. 2015). In the analysis, the FRAPTRAN-GENFLO fuel behavior 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.

As a part of the EPR LOCA analysis (Arkoma et al. 2015), the applicability of neural networks to predict the number of failing fuel rods was tested. A network was trained using the 1000 simulations from the worst global scenario. The network predictions were then compared to FRAPTRAN-GENFLO simulations results of all the rods in the reactor. It was discovered that the rods calculated by FRAPTRAN-GENFLO to survive were well predicted by the network to survive. However, a significant number of rods calculated to fail were not correctly predicted by the network. This can be understood by the fact that the number of failing rods used in teaching the network was very limited.

An alternative way to produce predictions from the existing data is to apply support vector machines (SVMs). SVMs have some advantages over neural networks; in this analysis, better performance in classification of the rods into failed/non-failed is expected. The analysis by SVMs has been initiated in 2016, 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 done in 2015 (Arkoma and Ikonen 2016a), helps in selecting the relevant input variables used for fitting the SVM. In the second phase, the fitted SVM is used for producing more data on the failed rods: the SVM is applied with new sampled input parameter values so many times that 1000 rods are predicted by the SVM to fail. In the third phase, FRAPTRAN-GENFLO simulations are made using the input values of these 1000 cases. Now we have 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.

The SVM's obtained in 2016 are then applied for each global scenario to capture the global effects. This part of the work starts in 2017 by investigating a single global scenario and evaluating the performance of the SVM's. The evaluation is done both against the statistical analysis of the full simulation data (Arkoma et al. 2015, Arkoma and Ikonen 2016b), and the sensitivity analysis done on the partial data in 2015-2016 (Arkoma and Ikonen 2016a,b). In 2018, the work is extended in to the full array of global simulation data and its sensitivity analysis.

The work done in 2017 will be reported as a research report or, optionally, as a conference or journal paper.

### 2.1.2 Methodology for determining input uncertainties (T1.2)

The task T1.2 started in 2016. The literature review of the potential methods will be prepared by the end of January 2017. In the review, existing methods used to quantify the uncertainty in thermal hydraulics codes are studied. In addition, methods used in other disciplines will be considered and their applicability to nuclear safety codes are studied.

In 2017, the approaches identified as the most promising are applied to a simple proof-of-concept case to test their applicability and results. Their performance is directly evaluated against methods that have been previously used, and reflected against experiences obtained in previous studies (Älku, 2013).

The work of 2017 consists of construction of a test model and mock-up experimental data for which the uncertainties are known, coding the methods under consideration in for example Python programming language and applying the methods to the test model. The methods used in previous studies are also applied to the test model and mock-up data for comparison of the new methods. Based on the findings the applicability of the new methods with Apros are weighted against the previously used methods. The work is reported in a research report.

### 2.1.3 Collision history-based GPT capability for Serpent (T1.3)

Task T1.3 is a new task starting in 2017 that will lay the groundwork for continuing two of the research directions of USVA that were put on hold due to the loss of multiple experts (Pusa, Isotalo, Tanskanen and Vanhanen) from the project group. Previously, the research into generalized perturbation theory (GPT) based sensitivity/uncertainty (S/U) calculations utilized a modified version of CASMO-4 (Pusa 2012a, 2012b, 2013, 2014).

Recently a novel collision history based methodology for GPT based S/U calculations was developed by M. Aufiero using an extended version of the Serpent Monte Carlo code (Aufiero 2015). The methodology is based on collecting collision history data for neutrons over multiple generations during a single Monte Carlo neutron transport simulation and calculating the requested sensitivity coefficients as a post processing step through a series of weight perturbations that are applied to the collision history. Due to the collection of history data through multiple neutron generations, the methodology is able to capture both the direct (resulting from changes in perturbed parameter) and indirect (resulting from changes in flux or adjoint flux due to changes in perturbed parameter) contributions.

The methodology has been successfully applied to the calculation of sensitivities of various result parameters such as reaction rates, time-constants, reactivity coefficients and k-effective due to perturbations in cross sections, material densities, scattering distributions and fission spectrum (Aufiero 2015, Kodeli 2016, Aufiero 2016a, 2016b).

For future applications in uncertainty propagation concerning burnup calculations, the methodology makes it possible to calculate the sensitivities of the fission and transmutation rates of specific nuclides to various input parameters including nuclide concentrations.

Regarding uncertainty propagation concerning multi-physics calculations, the methodology makes it possible to calculate the sensitivities of various output parameters, such as the power distribution to local changes in material densities or temperatures.

During 2017, the work in this task will focus on four simple subtasks:

1) Educating one new expert in the field of sensitivity/uncertainty calculations and uncertainty propagation.

2) Implementing and optimizing the collision history based methodology for S/U calculations into the official version of Serpent.

3) Applying the new methodology to assembly level sensitivity calculations in the context of the UAM benchmark.

4) Presentation of the implemented and optimized methodology as well as the results of the test calculations at the OECD/NEA LWR UAM-11 benchmark workshop.

Aufiero's implementation of the GPT methodology is based on the 2.1.19 version of Serpent publicly distributed in the March of 2014. During the porting of the methodology to the current publicly distributed version of Serpent,
the implementation will be optimized. The demonstration of the implemented methodology in assembly level calculations in the context of the UAM benchmark serves as an excellent verification of the implementation against previously calculated results with other tools (for example TSUNAMI and CASMO-4 in Pusa 2012b).

The work of task T1.3 will continue in 2018, when the application of the newly implemented methodology to coupled multi-physics calculations and/or burnup calculations will be demonstrated.

### 2.2 Multiphysics and calculation chain (WP2)

Work Package 2, Multiphysics and the calculation chain, concentrates on problems arising in coupled code simulations and calculation sequences. Code coupling can be achieved concurrently, where the codes are executed simultaneously and the solution is found self-consistently, or hierarchically, where the output of one code is used as an input of another, forming a sequence of calculations.

Previously in WP2, problems arising specifically in multiphysics simulations have been studied in collaboration with Aalto University. The work started in 2015 with a study of the coupled uncertainties in multiphysics simulations involving reactor physics and fuel behavior. The research was carried out as a special assignment and Master’s thesis work at Aalto University. In 2017, after personnel changes at the Aalto University, the task involves participation in a conference and reporting the results obtained in 2015-2016. The role of the task in 2018 will be evaluated during 2017. However, it is foreseen that the results obtained in Task 1.3 with the sensitivity analysis capabilities of Serpent will be extended to analyse Serpent-FINIX multiphysics simulations. Such work would be a direct continuation of the work done in 2015-2016.

In 2017, in the sole task of the work package (T2.1), the uncertainties related coupled reactor dynamics – fuel behaviour simulations are studied. The behaviour of the reactor during a transient depends greatly on the thermal reactivity feedback and the heat transfer from the fuel to the coolant. The behaviour is also very different for fresh fuel and previously irradiated fuel with moderate or high burnup. In 2017, the burnup initialization – the calculation of the initial state of the simulation – is studied. Although such a calculation can be done with a coupled fuel behaviour – reactor dynamics code system, alternative methodologies are also examined. The work of 2017 is in preparation for the year 2018, when the uncertainty propagation in such burnup initialization methods is studied.

Partners and person months allocated to WP2 are given in Table 2.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.0</td>
</tr>
</tbody>
</table>

### 2.2.1 Coupled calculations with fuel performance and reactor dynamics codes (T2.1)

As part of other the previous SAFIR2014 projects PALAMA and KOURA, and the SAFIR2018 projects PANCHO and SADE, the reactor dynamics and fuel behaviour codes developed at VTT have been integrated and tested. The results have been published previously, most recently by Ikonen et al (2015a, 2016). The next step towards uncertainty quantification in the coupled code system is the development of a methodology to take into account the burnup dependency in the initial state of the transient. This development has been started earlier in 2015 (partially outside of the SAFIR2018 programme), and will be finalised in 2017. The work is connected in part to Task T2.2, where the role of model approximation in determining the state of the gap and the gap conductance is studied in a broader context. The work of task T2.3 will continue in 2018, where the statistical uncertainties in the system developed in 2017 will be studied.

The work of 2017 in task T2.3 will be reported either as a research report or as research paper.

### 2.3 Administration and international collaboration (WP3)

Work Package 3 consists of project management and administration. In addition, travel expenses including the participation in conferences and relevant benchmarks are allocated to this work package. In 2017, the following participations are planned:

Possible travelling costs from participating in reference group meetings and other meetings requested by the SAFIR organization are also covered from WP3.

Partners and person months allocated to WP3 are given in the table 3.

Table 3 Partners and person months in WP3

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.0</td>
</tr>
</tbody>
</table>
3. Deliverables and milestones 2017

The deliverables and milestones for each task of USVA in 2017 are shown in Table 4. The milestones of the project are typed in the table in boldface.

Table 4 Deliverables and milestones in 2017.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Research report on the application of SVM’s to statistical LOCA simulations</td>
<td>2.5</td>
<td>End of January 2018</td>
</tr>
<tr>
<td>D1.2.1</td>
<td>Research report on the application of the input uncertainty determination methodology</td>
<td>2.5</td>
<td>End of June 2017</td>
</tr>
<tr>
<td>D1.3.1</td>
<td>Conference presentation at the UAM workshop</td>
<td>2.5</td>
<td>End of June 2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Research report or paper on the burnup initialization method in reactor dynamics calculations</td>
<td>1.0</td>
<td>End of January 2018</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Reports requested by SAFIR2018 administration</td>
<td>1.0</td>
<td>End of January 2018</td>
</tr>
<tr>
<td><strong>Total pm</strong></td>
<td></td>
<td><strong>9.5</strong></td>
<td></td>
</tr>
</tbody>
</table>
4. Project organisation

The USVA project is led by VTT Technical Research Centre of Finland. In 2017, Ville Valtavirta (VTT) will take over the role of the the project manager of USVA. Mr. Asko Arkoma (VTT) will be the deputy project manager. The project members work in areas of reactor dynamics (Elina Syrjälähti, VTT), fuel behavior (Asko Arkoma, VTT), reactor physics (Ville Valtavirta, VTT) and nuclear power plant behavior (Torsti Alku, VTT). The uncertainty analysis experts at Fortum (Timo Toppila, Karo Kustonen) have also been engaged in discussions with USVA. Dr. Risto Vanhanen and Mr. Aapo Taavitsainen (TVO, formerly of Aalto Univ.) are collaborating independently within task T2.1.

The project members, their participation by tasks and the estimated person months for 2017 are summarized in Table 5.

Table 5 USVA project members and their participation in tasks in 2017

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Asko Arkoma</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1, T3.1</td>
<td>2.75</td>
</tr>
<tr>
<td>Torsti Alku</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.2</td>
<td>2.5</td>
</tr>
<tr>
<td>Elina Syrjälähti</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T2.1</td>
<td>1.0</td>
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<tr>
<td>Ville Valtavirta</td>
<td>Project manager, Research</td>
<td>VTT</td>
<td>T1.3, T3.1</td>
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</tr>
<tr>
<td></td>
<td>Scientist</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
5. Risk management

Beyond the applied VYR funding and the contributions of VTT, USVA does not rely on any external funding. Therefore there is no excess risk related to project funding. For personnel resources, the current state of the nuclear energy field in Finland may imply a risk of increased competition for experienced personnel. This is a general risk in the field, and not specific to the USVA project. However, this risk is well acknowledged in USVA. To mitigate the risk, USVA has been actively seeking to educate new experts to the field, starting with a Master’s thesis project at Aalto University in 2015-2016. The education of experts is expected to continue in the following years of USVA.
References


Taavitsainen, A. “Coupled FINIX-DRAGON Calculation Chain for an LWR Pin Cell Case”, Special assignment in Physics, Aalto University, 2015.


Vanhanen, R. “Uncertainty analysis of infinite homogeneous lead and sodium cooled fast reactors at beginning of life”, Nuclear Engineering and Design, 283, Pages 168–174, 2015a


# The Finnish Research Programme on Nuclear Power Plant Safety (SAFIR2018)

## Resource Plan for 2017

**USVA**

Uncertainty and sensitivity analyses for reactor safety

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Material &amp; support</th>
<th>Travel</th>
<th>External services</th>
<th>Member fee</th>
<th>Other</th>
<th>TOTAL</th>
<th>VYR</th>
<th>VTT</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>person months</td>
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<td>85.8</td>
<td>59.7</td>
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<td>T1.1 Analysis of rod failures in LB-LOCA</td>
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<td>30.6</td>
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<td>T1.2 Methodology for determining input uncertainties</td>
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<td>30.6</td>
<td>21.3</td>
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<td>T1.3 Collision history-based GPT capabilities for Serpent</td>
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<td>WP2 - Multi-physics and the calculation chain</td>
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<td>T2.1 Coupled calculations with fuel performance and reactor dynamics codes</td>
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<td>8.5</td>
<td>3.7</td>
</tr>
<tr>
<td>WP3 - Administration</td>
<td>1.0</td>
<td>12.0</td>
<td>0.0</td>
<td>5.0</td>
<td>0.0</td>
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<td>17.0</td>
<td>11.8</td>
<td>5.2</td>
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<td>T3.1 Project administration and international collaboration</td>
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<td></td>
<td></td>
<td>17.0</td>
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<td>5.2</td>
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<tr>
<td><strong>TOTAL</strong></td>
<td>9.0</td>
<td>103.7</td>
<td>6.3</td>
<td>5.0</td>
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<td>0.0</td>
<td>0.0</td>
<td>115.0</td>
<td>80.0</td>
<td>35.0</td>
</tr>
</tbody>
</table>

Comments:

Matt and supp include computing cluster costs at VTT (790 €/pm).

Travel includes 1 conference participation (Mathematics and Computation 2017) and a workshop participation (UAM-11), and required project meetings (such as RG).
Steering Group SG3 –
Structural safety and materials:
SAFIR2018 Project plan

**COMRADE**

Condition Monitoring, Thermal and Radiation Degradation of Polymers Inside NPP Containments

Konsta Sipilä¹, Sami Penttilä¹, Harri Joki¹, Antti Paajanen¹, Tiina Lavonen¹, Marcus Granlund², Anna Jansson², Anna Bondesson², Jan-Henrik Sällström², Johan Sandström²

¹ VTT Technical Research Center of Finland

² SP Technical Research Institute of Sweden
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   1.1.2 Synergy of radiation and heat and DLO effects ..............6
   1.1.3 Modelling ............................................................6
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1. Research theme and motivation

A joint Nordic project on Condition Monitoring, thermal and Radiation Degradation of polymers inside NPP containments (COMRADE) has been developed, following an initiative from the Nordic NPPs through Energiforsk. The project development has been made by SP Technical Research Institute of Sweden and at VTT Technical Research Center of Finland.

COMRADE is developed based on input from a feasibility studies from Energiforsk AB [2015:157] and STUK [Penttilä, 2016] 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. The plan is to divide the work into different steps, all with the aim of providing the power plant operators as well as regulators and polymer manufactures with a deeper knowledge of the degradation of polymers and to develop methods for setting acceptance criteria of polymeric materials. This is done through three work packages:

- WP 1 focusing on method development of condition monitoring and implementation at NPPs
- WP 2 is a pre study to map how closed down power plants, such as Barsebäck, Oskarshamn 1 and materials taken from outages can be used to verify the method developed in WP1 and degradation process studied in WP3.
- WP3 focusing on polymer ageing mechanisms and effects inside the NPP containment and ageing management of polymeric components

An additional WP4 is included in the project that focuses on the improvement of international cooperation in the field of polymer component ageing and ageing management between the project team and Nordic stakeholders. Aim is also to improve the knowhow related to polymer ageing inside NPP containments by contacting and involving international experts of this specific topic.

By completing these work packages the nuclear power plants and radiation safety authorities will have the possibility to use functional based acceptance criteria, see how materials available from Barsebäck and currently running plants can be used to verify models for acceptance criterias, gain better understanding in synergistics effects between heat and radiation and see how different levels of low dose gamma radiation will affect the polymeric component. Also the current practices related on ageing management of different polymer components are provided.

The work will be done in cooperation between VTT, SP, the Swedish and Finnish Nuclear Industry through Energiforsk, Nordic regulators and a manufacturer of nuclear grade elastomers. Polymer experts from the Nordic NPPs will have an active role in all WPs The work packages are all managed by one of the institutes, but the work done within the packages is shared between the participants.

1.1 Background and state-of-the-art

According to IAEA, operating experience has shown that ineffective control of ageing degradation of the major Nuclear Power Plant (NPP) components can jeopardize plant safety and also plant life. Ageing in NPPs must therefore be effectively managed to ensure the availability of design functions throughout the plant service life [IAEA, 2000&2012]. As discussed at the Energiforsk (ELFORSK) Seminar on aging of polymeric materials in
NPPs held in 2014 the importance of defining a method to determine the acceptance criteria for polymeric components was pointed out. The acceptance criteria should be based on functional demands since it is the first step in the process of lifetime estimations of existing or new materials. As a result of the seminar a feasibility study featuring interviews with polymer experts at the NPPs in Sweden and Finland was launched to investigate the preliminary viability of defining a method to determine acceptance criteria for polymeric materials. The feasibility study was carried out by polymer experts at SP during March through August. The study started with a set of interviews with the 5 Nordic nuclear power plants Loviisa, TVO, Forsmark, OKG and Ringhals. Through the interviews a chosen set of components were identified to be included in the feasibility study. This includes but is not limited to type of polymer and environmental aspects such as heat and radiation. Based on the components, and the type of polymers used for it, a literature study was done in order to identify existing knowledge on aging mechanisms. From the knowledge gained through the literature study and existing know-how at SP, tables with suitable methods was proposed, if possible, acceptance criteria was concluded for each component. However, it was difficult to find already defined acceptance criteria for the proposed polymeric components.

As concluded in the feasibility study [2015:157] there are issues with identifying already available acceptance criterias based on functional demands in existing research. There are tests done when material properties have been monitored during accelerated ageing through both radiation and heat. There is one recent doctoral thesis done at KTH Polymeric materials in nuclear power plants – Lifetime prediction, condition monitoring and simulation of ageing [Linde, 2015] which tests how properties changes over time in accelerated tests using both radiation and heat. The test was made on insulation for electrical cables (PVC, EPR, XLPE and NBR). Even though the tests showed interesting results for evaluating ageing properties tests done are not verified by the actual function of the polymeric component but focuses instead at the degradation of a material property. To be able to set an acceptance criteria based on the function, there is a need to correlate the material property changes to a change in the function of the polymeric component.

VTT conducted feasibility study on ionizing radiation environments inside NPP containments and irradiation resistance of used polymer components for STUK during 2015 and 2016 [Penttilä et al. 2016]. During the study the material and radiation experts working on Finnish NPPs were interviewed. Data on the most commonly used polymer grades used inside NPP containments and ionizing radiation environments during normal service life, service failures and severe accidents were gathered. The ageing environment for a polymeric component can vary very significantly during each scenario and thus the functionality of the polymer component needs to be taken into account when the acceptance criterion for the component is considered. The set acceptance criterion should be some functional property of the component (e.g. leak tightness in case of O-rings) or some property that could be correlated to it with sufficient accuracy. Clear effects rising from the varying dose rate were confirmed even though all polymer types do not show clear dose rate effects. Also more detailed understanding from the synergistic effects of radiation and heat were recognized to be required in order to define more accurate accelerated ageing procedures to be used during material qualification.

### 1.1.1 Ageing

Based on the interviews in the Energiforsk report 2015:157 different components where chosen by the 5 Nordic NPPs. These components are estimated to cover a large portion of the polymeric materials in system components at an NPP and a more detailed percentage is provided in WP1. By studying function based acceptance criteria for these components and correlating it to a material property the NPPs will be able to better understand the lifetime of their polymeric components. It will also help the regulatory authority to better understand how the function of the polymeric components changes over time and what causes the degradation.

Polymeric materials used in NPPs are subjected to different temperatures, dose rate and total doses of ionizing radiation. The polymer’s expected life time depends on its resistance to degradation and the environment it is subjected to. During its life time it will receive a total dose of radiation at a specific temperature. These are factors that need to be taken into account when developing the method for ageing. As written in Prediction of service lifetimes of elastomeric seals during radiation ageing [Burnay, 1984] historically radiation tests on polymeric material, samples have been performed at high dose rate ~10 kGy/h (1 Gy = 100 rad) to obtain “lifetime dose”. However, it has been well established that many polymers exhibit dose rate effects or synergism between temperature and radiation making high dose rate predictions of limited use. An exposure at high dose rate radiation also concentrates the ageing to the surface, which can generate insufficient correlation to real use. For accelerated ageing using temperature the relationship between temperature and heat may follow the Arrhenius equation which is commonly used for accelerated tests for polymers not subjected to radiation. It is important for both heat and
radiation to consider the potential risk in changing the degrading reactions when using too high temperature or too high dose rate.

The dose rate for the components, during normal operation, in [2015:157, Penttilä et al. 2016] ranges from none, low (mGy/h) to a highest range of (0.1 – 1.0 Gy/h). These dose rates are far below what has been used in experimental trials identified in the literature study. Values are often in the magnitude of 200 times higher than what is estimated as the higher dose rate range for the components in [2015:157]. Therefore, a study in WP1 will be done using lower dose rates for the work done when studying the functional based acceptance criterias. Radiation induced ageing during DBA are studied in WP3 in which higher dose rates are used with or without coupling them to thermal stresses.

The existence of dose rate effect is evident from the data that is available from the open literature. Dose rate effects can be observed mostly at high or very low dose rates. When intermediate dose rates are considered, existence of dose rate effect is rare. The dose rate effect seen at high dose rates is due to diffusion-limited oxidation (DLO) and a separate task within this project is concentrated solely on this topic. At low dose rates the magnitude of dose rate effect is governed by which polymer related thermal or radiation degradation reaction pathways are dominant [Gilien et al. 1993]. Some polymers are known to be more susceptible to dose rate effect than others. However, real data obtained with very low dose rate is sparse and the dose rate effect observed during accelerated irradiation ageing treatments is generally due to DLO. Thus instead of using low dose rate experimental data, dose rate effects can be estimated with semi-empirical models that use experimental data obtained at relative high dose rates and possibly thermal ageing data, in order to extrapolate the material degradation to lower dose rates [IEC 61244-2].

1.1.2 Synergy of radiation and heat and DLO effects

The synergism in degradation of a polymer when exposed to heat and radiation is further supported in Degradation of elastomer by heat and/or radiation [Masayaki, 2007] for EPDM and at 140/70 °C and 5 kGy/h / 3 kGy/h. The test showed synergism for the degradation with an increase in rate when the material was subject to both factors. According to Kuriyama et al [Kuriyama, 1979], the value of $E_A$ (activation energy) for nonirradiated samples is typical of thermo-oxidative degradation process, however for irradiated samples the values of $E_A$ are more typical of oxygen diffusion in polymers. In the referred tests the oxygen penetration of the cross-linked polyethylene was limited to the surface (~0.6 mm at dose rate of 0.1 kGy/h). Synergy effects during aging treatments are not included in regulatory instructors [Häkkä-Rönnholm, 2004] and evaluation of this effect is further clarified with literature survey and experimental work. Whether these synergy effects should be noticed when creating acceptance criteria for different components is determined.

Due to the morphology of the material, distribution of antioxidants, oxygen availability etc. ageing is a heterogeneous process. Diffusion of oxygen is often a rate determining step and surface degradation is observed in several studies. Decrease in mechanical properties such as elongation at break is often observed as a consequence of surface degradation [Wündrich, 1985]. Most incidents caused by failure of polymeric materials subject to radiation are related to elastomeric seals and electric cables. Even though many studies show a synergy between heat and radiation most of the failures are caused by thermo oxidative ageing rather than radiation ageing [Kotthoff, 1994]. However, it is still unclear how the diffusion of oxygen effect to the mechanical properties of polymers in NPP environments. There is no clear opinion how much the surface defects induced during heterogeneous oxidation decrease mechanical properties compared to homogeneously oxidated material.

1.1.3 Modelling

SP Structural and Solid Mechanics has performed a research project on leak tightness in large polyethylene pipe joints with rubber gaskets [Jacobsson, 2011 Part 1 and 2]. The project included modelling leakage with finite elements using creep models in the polyethylene and hyperelastic modelling of the rubber, followed by full scale leak tests on the joint. The project is now followed by a new ongoing research project. This current project features material testing to calibrate elastic- plastic creep modelling of the polyethylene and hyperelastic nonlinear viscoelastic modelling of the rubber used in steel reinforced o-rings. The material models are used in a full model of the joint with fluid penetration contacts to simulate leakage and accompanied by full scale leak tests.
Previous research also includes evaluation of the fatigue life of rubber components, where FE-modelling in combination with material testing was compared with other fatigue life evaluation methods (fracture mechanical approach and cracking energy density approach). Fatigue testing was then performed on the actual rubber components compared with the results of the fatigue life evaluation methods. Other, previous or ongoing, research includes determination of mechanical properties of rubber and aging of seals in water pipes.

When determining mechanical properties of rubber, a set of tests has been performed in the laboratory at the assumed future operating temperature. The set consists of uniaxial tension and compression, and also pure shear strain test and volumetric compression.

A general investigation was done on water tightness of rubber seals for the Swedish Water and Waste Water industry [Thörnblom, 2014]. It was concluded that the required ageing tests in the applicable standards are far from sufficient. The tests performed cannot be useful for forecasting a life of about 100 years, which lies in the area of the networks administrators need. An extensive study on ageing of seals (EPDM, NBR, TPE) is proposed to the Swedish Water & Wastewater industry.

1.1.4 Results of 2016

WP1: During 2016 in task T1.1 peroxide cured EPDM o-rings at standard cord size 3.53 mm was provided by James walker for testing. Test blocks for tightness test was manufactured by SP and adapted to the o-ring size to be able to receive the correct compression. The test run was started with radiation treatment for two weeks followed by two months of heat treatment at three different temperatures. Both o-rings and dogbones (same material) is used in the ageing process. In parallel to the samples treated by radiation the same set of samples is ging through heat treatment. The starting point and the first evaluation point has been completed and presented during the workshop in September. Adjustments to the ongoing tests have not been made and the second and final radiation treatment has been completed. For the upcoming tests a larger cord diameter will be used. A larger number of samples to be able to run relaxation are also considerd. For the implementation phase, the use of XRD-analysis was tested in analysing the sulphur content from the sulphur cured o-rings. Referene measurements on peroxide cured o-rings were also conducted in order to evaluate whether the method could detect the sulphur content from the o-rings.

WP2: In the contact established with the Barsebäck plant they informed that they are able to provide polymer materials with documentation. Their proposal was to select a certain small pump inside the reactor containment. This pump has a plastic impeller and o-rings. The pump has been serviced yearly so the parts are not very old. They have not found any candidate joint sealants or cable seals (Brattbergare) that have been exposed to radia- tion. If any exist they are not well documented. Barsebäck plant does not have the ability to give radiological clearance in situ, and it is not certain that the equipment that has been in the enclosure can be given radiological clearance at all. If it cannot be given radiological clearance the examination must be done in the controlled area. If the investigation will be done in situ the following things must be arranged: admission, training, dosimeters and our equipment will have to be given radiological clearance afterwards. This would yield in significant cost. Another aspect of taking material from Barsebäck is that after the outtake of the reactors the materials have been stored for many years in different temperatures and atmosphere than when in use. Because of these difficulties the search for radiated polymeric materials will be broaden to include the NPPs in service and plants that just have been or soon will be taken out of service.

WP3: The mechanisms behind synergism can be very complicated, involving a plethora of chemical and physical processes across multiple structural and time scales. Current state-of-the-art in lifetime prediction involves semi-empirical methods based on fitting analytical models againts data obtained from accelerated ageing tests. Despite all the progress in the fields of computational materials physics and chemistry, there remains a considerable gap between the present multi-scale materials modelling capabilities and practical lifetime prediction and bridging the gap would require remarkably large resources. Thus the future work should be concentrated to modelling of a relevant detail of the ageing process which the current methods fail to take into account causing faulty estimations on the remaining lifetime. The reverse temperature effect on semicrystalline polymers was recognized to be such problematic ageing process that is needed to be understood in more detail in order to produce improved lifetime prediction methods.
During 2016 the synergistic effect of radiation and heat were analysed on EPDM and Lipalon cable jacketing materials by means of tensile testing and hardness measurements. The material testing and analysis is currently ongoing and the first experimental results on EPDM tend to show that simultaneous exposure to heat (T=125°C) and radiation is less degrading to EPDM when compared to situation where EPDM is solely irradiated at room temperature. Complete report on the synergistic effects will be available on December 2016.

The concentration of oxygen on the surface and in the bulk material was compared by using ToF-SIMS (Time of Flight Secondary Ion Mass Spectroscopy), FTIR (Fourier Transform Infrared spectroscopy) and DSC (Differential Scanning Calorimetry). The ToF SIMS seemed to produce very detailed data on oxygen content of the sample surfaces studied. According to ToF SIMS measurements, the oxygen content is actually more concentrated to the surface in the control samples (no irradiation treatment) than the irradiated sample. More samples need to be analyzed after different radiation and thermal exposures in order to better understand the degradation processes. Also different analyses must be compared to verify the results. Complete report on results will be available on December 2016.

The dose rate effect does indeed exist on most polymers at very high dose rates and very low dose rates. At intermediate dose rates the existence of dose rate effect is very rare. At high dose rates, dose rate effect yields from diffusion limited oxidation. The magnitude of dose rate effect on the very low dose rates depend on on the governing thermal and radiation degradation pathways i.e. material specific reaction kinetics. There are few semi-empirical methods that can be used in extrapolation of dose rate effect based on e.g. superpositioning. However, these methods require large amount of experimental data (both on thermally and radiation aged samples) and their use is limited to certain type of polymer types. They are reliable to predict dose rate effects on dose rates a decade smaller than the dose rates used in the ageing data that the model uses. On further dose rate extrapolations uncertainty is increased.

1.2 Objectives and expected results

This project consists of three main objectives which include improving condition monitoring of polymeric components used inside containments, providing ageing data which is used in evaluation of acceptance criteria for polymer components and providing tools for robust ageing management for these components. To achieve these objectives three different work packages have been created which study polymer ageing in different perspectives.

- WP 1 focusing on method development of condition monitoring and implementation at NPPs
- WP 2 is a pre study to map how closed down power plants, such as for example Barsebäck, Oskarshamn 1, and materials taken from outages in running NPPs can be used to verify the method developed in WP1 and degradation process studied in WP3
- WP3 focusing on polymer ageing mechanisms and effects inside the NPP containment and ageing management of these components
- WP4 improves international cooperation on polymer ageing issues at NPPs between Nordic researchers, NPP operators and regulators

These work packages are based on issues recognized during feasibility studies on acceptance criteria of polymer components conducted in Finland and Sweden [2015:157, Penttilä et al. 2016].

The objective of work package 1 is to identify the acceptance criteria for the function of the polymeric component. This includes

- Development of test methods
- Performing experimental tests to validate the method
- Development of a theoretical model that can be used to calculate different geometries
- Deployment of the results into the daily operations at the NPPs.

The polymeric component that will be tested is an o-ring and there will be 3 different materials tested; EPDM, Nitrile and Silicone or Viton. The size of the o-rings is decided based on available components and the possibility to fit the test blocks in the gamma cell. A cord diameter of 3,5 mm is used for the first test with EPDM but a cord size of approximately 7 mm will be used for the remaining part of the project. This is due to avoiding to much surface effects. The implementation will be done in close collaboration with the interest group.
The objective in work package 2 is to fulfil a pre-study for polymeric materials from closed down power plants, such as Barsebäck, Oskarshamn 1, and materials taken from outages in running NPPs that have undergone ageing for many years. This will include: identify the polymeric components that can be available to study, the amount of data that exists on these today and the access of these at the plant. The pre study will also include a workshop to present and discuss the results.

Studying the combined effects of radiation and temperature in work package 3 is performed in the form of both theoretical and experimental work. The literature survey carried out in 2016 focused on the mechanisms underlying the synergistic effects, and on possible ways of modelling them. It was concluded that the mechanisms behind the synergism can be remarkably complicated, and that there is a considerable gap between the present multi-scale materials modelling capabilities and practical lifetime prediction methods. Nevertheless, many key processes were identified, and most of them could be coupled with a suitable modelling approach. The remainder of the theoretical effort was decided to be directed in explaining the “reverse temperature effect”, an anomalous phenomenon observed in certain semi-crystalline polymers. This reverse temperature effect complicates the predictability of polymer ageing behaviour and the current lifetime prediction models tend to significantly underestimate the ageing degradation for these materials [Burnay&Dawson 1999, Celina et al. 1996]. In order to make reasonably reliable predictions on lifetime of semi-crystalline polymer components, better understanding on the reverse temperature effect is required. In the experimental section of year 2016, the amount of degradation caused by the synergy was determined with a test matrix that compares the synergy induced degradation to radiation and thermal degradation only. Complementary analyses for the samples are done during 2017 to identify the degrading structural areas within the samples. This will provide data that is required when synergistic effects during DBA are evaluated.

During 2016, the effect of oxidation depth was analysed with different methods (ToF-SIMS, FTIR, DSC) and tensile testing in order to determine detrimental effects of surface oxidation and bulk oxidation on samples that were irradiated with a constant dose rate. In 2017 the DLO effects are studied with samples irradiated with different dose rates. This section provides also data of material behaviour during a service failure and the same data is needed when the lifetime prediction model is applied on dose rate effect extrapolations.

Dose rate effect studies aim to clarify whether lower dose rates tend to cause significant amounts of degradation to polymer components compared to high dose rates during DBA and simulated normal service lifetime ageing treatments. Since irradiation treatments with low dose rates are very time consuming and costly, more theoretical approach is adopted as different extrapolation methods are studied. Extrapolation methods were evaluated by their applicability during 2016, and based on the evaluation one or two suitable methods are used in evaluation of dose rate effect. During 2017, experimental data is produced and used in evaluation of dose rate effect in relatively high dose rates (ca. >60 Gy/h) and same results can be used as input data for the chosen extrapolation method, in order to evaluate the severity of dose rate effect during normal service lifetime and its relation to accelerated ageing.

Functioning ageing management programmes are essential in safely operated NPPs. Especially for plants that are getting close to their original designed end-of-life and evaluating a possible lifetime extension. The current applied on-site procedures related to ageing management of polymer components and the amount of ageing issues can differ between plants. Due to this, the current status of ageing management of polymer components (cables, sealants, paint coatings, lubricants and greases) is summarized and the standard ageing management procedures related to them are gathered in the state-of-the-art survey. This will provide valuable information (e.g. lifecycle of different components) for nuclear consortium on the current state of ageing management and will act as basis when improvements to ageing management procedures are developed.

1.3 Exploitation of the results

Challenges in the polymeric materials, such as cable materials, O-rings, joint sealants and linings are recognized in the current power plants as well as in the new plants. In this project, strong international co-operation will be carried out for obtaining new information and improving the expertise on polymeric materials in nuclear applications in the Nordic countries. The ultimate end users of the project results are the power plant operators as well as regulators and polymer manufactures, which are involved with assessing the integrity of the components that may be subjected to ageing degradation by radiation and heat.
Understanding the combined effects of ionizing radiation and temperature is focal when evaluating the lifetime of polymeric components during normal service life or in DBA situations. The potential synergistic, antagonistic and competitive effects of these environmental stressors should be considered, e.g., when evaluating acceptance criteria for new components. On the theoretical front, atomistic simulation methods provide one way to complement the existing experimental knowledge. The modelling work carried out in COMRADE will provide mechanistic insight into the coupled elementary processes of thermal-radiative ageing. The generated knowledge will be especially useful when considering materials that display the so-called reverse temperature effect. Furthermore, the work constitutes a step towards mechanistic kinetic modelling of thermal-radiative ageing, a modelling regime which could be developed for materials that are problematic from the viewpoint of conventional lifetime prediction methods.

Experimental work on the synergistic effects and dose rate effect will contribute to better understanding of polymer ageing on realistic and accelerated conditions. The qualification processes address currently only high dose rates to be used during the ageing procedure. Within this project, data is obtained at different temperatures, dose rates and absorbed doses. The data can be used to evaluate if currently used accelerated ageing procedures sufficiently simulate reality. Based on the results, more realistic accelerated ageing procedures can be developed improving the qualification of nuclear grade polymer components.

Furthermore, the results gained from the project will allow regulators, power plant operators and polymer manufacturers to work with polymeric materials with greater knowledge concerning ageing phenomena and acceptance criterias. This will allow improved monitoring of polymeric materials, life time prediction and to make sure the component is replaced at the correct time. This will also help estimating the status of a component before and during accident conditions. The ageing management procedures can be updated at plants resulting in improvement in nuclear safety and cost efficient component maintenance. The project contains tasks for implementation in the work packages which means that after the implementation phase it is estimated that the knowledge, test method or other results can be used by the regulator, power plant operators and polymer manufacturers.

An interest group called advisory group was formed during the first year of the project which includes representatives from NPPs, authorities and a supplier. The advisory group will participate with their knowledge on the needs of the end users and they will also be an important communication channel to the industry and the authorities. For instance, during the deployment phase in WP1, the group will act as the end users of the developed method. There will be annual seminars for advisory and project groups where project content and results are presented and discussed.

The results gained from the project will also strengthen the regulatory authorities’ competence concerning ageing phenomena and acceptance criteria of polymeric materials in their role as a supervisory authority. It will allow improved monitoring of polymeric materials, life time prediction and to make sure that components are replaced at the correct time. This will also help estimating the status of a component before and during accident conditions.

The direction of the proposed method development is relevant for ageing control, which is essential for long term operation of nuclear power plants. It also strengthens the competence within the regulatory authorities. Openly published scientific-based results will increase the knowledge base with regards to ageing of polymeric materials in environments with ionizing radiation. Given the active involvement of the nuclear power plants, the results will also be implemented in the daily operation of the nuclear power plants.
<table>
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<th>End users</th>
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<td>Study ageing effects on functional demands of o-rings</td>
<td>Acceptance criteria for o-rings</td>
<td>NPP operators, regulator</td>
<td>2018</td>
</tr>
<tr>
<td>Development of FE-model for o-rings</td>
<td>Use of experimental data in model parameter calculations</td>
<td>Modelling tool for o-ring performance</td>
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<tr>
<td>Study materials available from OKG, Barsebäck and other plants during outage</td>
<td>Identify available components for study and gather detailed data on these components</td>
<td>Data set that can be further used when ageing during normal service life is evaluated</td>
<td>NPP operators, regulator, component manufacturers</td>
<td>2017</td>
</tr>
<tr>
<td>Evaluate the most important stressors and ageing mechanism governing the ageing during normal use and DBAs</td>
<td>Gather data from NPPs and provide experimental data to evaluate severity of dose rate effect and synergistic ageing effects</td>
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<td>NPP operators, regulator</td>
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<td>Development of molecular dynamics models for studying synergistic effects in thermal-radiative ageing</td>
<td>Identify the primary molecular-level processes behind the reverse temperature effect</td>
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<tr>
<td>Improve ageing management of polymeric components</td>
<td>Interview NPP experts and identify current state of ageing management of polymer components and improving procedures.</td>
<td>State of the art on ageing management of polymer cables and possible suggestions for improvements.</td>
<td>NPP operators, regulator</td>
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<td>Improve the cooperation between Nordic NPP operators, regulators and research community on polymer component ageing issues</td>
<td>Annual workshop where project results and polymer component ageing issues are discussed</td>
<td>Gain of new knowledge and information exchange among the stakeholders and research team</td>
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1.4 **Appropriateness of the project to SAFIR2018 programme**

The SAFIR2018 programme emphasises new openings improving knowledge on NPP component ageing and lifetime management. In particular the framework plan cites irradiation resistance of cable materials as a new important field of research, and irradiation-induced ageing of organic materials is mentioned in the framework plan. The project studies the influence of irradiation and heat on a number of different polymer materials in the cases of design base accidents and normal service together with research institutes and polymer manufacturers, making it a cross-cutting project benefitting from the specific backgrounds of the experts in the Nordic countries. The project enables access to important research areas in the nuclear field e.g. study polymer components and ageing environment at currently operating NPPs and Barsebäck.

The ageing of polymeric materials has been studied to some extent but the simultaneous effect of radiation and heat is a less commonly studied topic. Typically, many polymeric materials are used in locations / components where they can be replaced. However, according to radiation safety principle ALARA, all reasonable methods must be employed for minimizing radiation doses to NPP personnel. Thus this safety principle supports strongly the use of ageing resistant materials in order to avoid unnecessary radiation doses to personnel. The role of ageing resistant materials in the improvement of reliability and safety must be taken into account by research since the importance of defining the acceptance criteria for polymeric components has been pointed out lately [2015:157, Penttilä et al. 2016]. The acceptance criteria should be based on functional demands since it is the first step in the process of lifetime estimations of existing or new materials.

Strong international cooperation between VTT, SP Technical Research Institute of Sweden, the NPPs and polymer manufacturers are foreseen within this project concerning the combined effects of radiation and thermal ageing on polymeric materials. This project supports the strategy of SAFIR2018 and brings improved capabilities for testing of different types of polymeric material groups under ionizing radiation. The work will start with o-rings in the first year but in the future for instance joint sealants and lining material are of interest. This project also features the adaptation of diverse techniques in testing and characterisation for use with irradiated polymeric materials, including the compression set, hardness, stress relaxation, DSC, FTIR and tensile testing as well as different modelling tools. Throughout the work packages synergism between experimental work and modelling can be seen. This will enhance the quality of scientific publications and provide better solutions to practice.

1.5 **Education of experts**

Staff from the NPPs will take an active role in the project, both junior and senior personnel will be invited to participate. The knowledge gained from the work packages will give the NPP experts better knowledge in the degradation of polymeric components. The knowledge concerning degradation of system components includes components used in safety classified equipments/functions. The possibility in setting acceptance criteria for safety classified components gives the plant a better possibility of having better control of the equipment through the intended lifetime including setting the requirements for new components. Dr. Sue Burnay from John Knott Associates Ltd. will act as a senior advisor and will provide her expertise for the project team. Dr. Burnay has 50 years of experience on polymer ageing issues in NPP environments so her view on project results is valuable and provides a possibility to the project team to improve their knowhow on this research area.

The intention of the increased knowledge in regards to acceptance criterias is to be able to better estimate when a component needs to be replaced, know what functional/material requirements to set when purchasing new components, and to avoid the component to fail earlier than expected. Furthermore, a better understanding on how low too high dose rates will affect polymers will be gained.

A master thesis is planned to be done together with Chalmers University of Technology Nuclear engineering and polymer technology department. The thesis work will be included in work package 1 lead by SP and is planned for spring 2017. The student will work with studying the ageing process to be able to better define the acceptance criteria for the function tightness of the o-ring.
Two workshops are planned together with the nuclear power plants and nuclear safety authorities. The first workshop was organized in Borås by SP during 21-22nd September 2016 and the second one will be organized by VTT during autumn of 2017. The purpose of the workshops is to present the most recent results from the project, present the next steps and educate the project team, NPP staff and regulators on related polymer ageing topic lectured by Dr. Sue Burnay. This will allow the end users to give their view on the ongoing work, the possibility to influence the way forward and improve their expertise.
2. Work plan

Table 2. The overall plan is shown in the below Gantt chart.

<table>
<thead>
<tr>
<th>Task</th>
<th>2017</th>
<th>2018</th>
<th>Comments</th>
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</thead>
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<td>T3.5</td>
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2.1 Work package 1 (WP1) Development of condition monitoring methods for polymeric components including low dose rate radiation exposure.

The aim of this work package is to identify the acceptance criteria for the function of the polymeric component. This includes:

- Develop robust test methods that can be used by the power plants for condition monitoring through a material property. The material property will be correlated to the function of the component.
- Performing experimental tests to validate the method.
- Development of a theoretical model that can be used to calculate acceptance criteria for components with different geometries.
- Deployment of the results into the daily operations at the NPPs.

During a feasibility study [2015:157] acceptance criteria for functional properties for different polymers in system components was studied. The components were selected based on interviews with the five Nordic nuclear power plants. Furthermore, the need to study degradation using low dose rates was identified since previous work described in literature mainly focuses on using a high dose rate to achieve the life time dose during a short period of time. This may cause a different degradation, compared to that obtained with a long exposure at a low dose rate.

The study will focus on accelerated ageing through heat and radiation to selected components starting with o-rings. Both Teflon seal and reinforced EPDM are of interest and are valid components to be tested in the future but left out of this work package. The test will be done using one or several specific properties, compression set being one of them, and correlate this to a functional test for tightness of an o-ring. By doing so a correlation between compression set and tightness can be achieved. To be able to better compare the effect the radiation has on degradation, a parallel test in heat will be done.

The o-rings will be mounted in tube connectors during exposure and dogbones will also be included, to be able to increase the number and properties to be tested. Compression set after certain exposure times will be measured and the pipe connections will be tested by mounting them in the SP hose testing equipment. With this test the sealing performance will be measured as water pressure without leakage.
By testing the correlation between compression set and tightness of an o-ring an understanding of the function (tightness) based on a material property (compression set) can be made. The aim is to be able to use this to set acceptance criteria for an o-ring using compression set as a property.

Other evaluation methods include tensile testing, thermal analysis (Oxidation Induction Time, DSC) and Nuclear Magnetic Resonance. DSC has been identified as a valuable method since antioxidant content is indirectly measured and indicates the residual service life of a product. Another benefit is that only small samples (5 mg) are needed. Since the knowledge about the material composition is often limited it is valuable to analyse composition (amount of plasticizer and filler), type of antioxidant and vulcanization system.

The following materials are to be included in WP1:

- O-rings (EPDM) Temperature* 90, 120, 140 °C Vulcanized and subject to low radiation.
- O-rings (Nitrile) Temperature* 60, 80, 100 °C, Vulcanized and subject to low radiation.
- O-rings (Silicone or Viton), Temperature* to be decided depending on the chosen polymer. 3 different temperatures to be used, Vulcanized and subject to low radiation.

*Temperature may vary slightly due to the formulation of the polymer. The temperature for the test will be set once the polymeric component has been provided.

The dose rate during irradiation exposure is set to 21 Gy/h during a total of 28 days. This gives a total dose of 14 000 Gy to be compared to a component during normal operation subjected to a high radiation environment of 0.1 Gy/h for a little bit over 16 years. In future work (2017 and onwards) ageing parameters can be chosen in a way that functional properties of these components can be evaluated in situations like service failures and severe accidents in order to provide acceptance criteria if the component has designed function during these situations.

To get an idea of how much of the polymeric materials used in the NPPs that would be covered using the results from this project, actual numbers from Ringhals o-rings (EPDM, Nitrile, Viton) indicate a coverage in order of 85-90% [Widestrand, 2015 email]. If choosing silicone instead of Viton the coverage will decrease but still is approximate 65-70% at Ringhals.

The timeline below followed by a test matrix, show the sequence of tests to be completed for one component including radiation. During the irradiation treatment the heat in the chamber will be approximately 25 °C. The test matrix shows 12 samples running at three different temperatures, with or without radiation and at two different geometries (cord diameter of the o-ring). Only the EPDM o-ring will be tested using two cord diameters. This is estimated to be enough for the modelling but more tests could be proposed as future work if determined to be of interest. There are 5 points for evaluation including starting point. The time between evaluations is decreased at the later stage of the test since it is the region where the acceptance criteria or end of life will be found. It is estimated that a minimum of 80% compressions set is needed before the function will fail. Higher then 80% has been achieved for half time evaluation for the samples running at 140 °C and the sealing performance is still working.

![Time line for 1 test including radiation and heat treatment. The temperature and radiation rate can be found in table 3. Dates are exemplified with start January 1st 2016, actual start see Deliverable chapter.](image-url)
Table 3. Test matrix for WP1 exemplified for EPDM. Geometry stands for diameter of o-ring.

<table>
<thead>
<tr>
<th>Sample</th>
<th>Temperature (C)</th>
<th>Dose rate (Gy/h)</th>
<th>Cord diameter (mm)</th>
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<tr>
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<td>21</td>
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</tr>
<tr>
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<td>21</td>
<td>6</td>
</tr>
</tbody>
</table>

Since there are many different sizes and shapes of o-rings used at a nuclear power plant, a theoretical model using the data identified through tests in WP1 will be created and verified. The model can be used to estimate acceptance criteria for a larger number of components with different geometries during a shorter time, than actual tests.

The result may be possible to use in technical documents setting requirements for polymeric components for the nuclear power plants. This can be used for existing components in the NPP or when purchasing new components (A fingerprint through for instance FTIR or DCS should be added). Depending on the components identified in WP2 a comparison can be made to the accelerated test in WP1.

The results will be presented at a seminar and in a report. Depending on the findings, a scientific article will be written and presented.

Partners and person months allocated to WP1 to be given in the table.

Table 4. Partners in WP1.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
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<tbody>
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<td>James Walker*</td>
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<td>Nordic NPPs*</td>
<td>0.2</td>
</tr>
<tr>
<td>John Knott Associates Ltd.</td>
<td>0.05</td>
</tr>
</tbody>
</table>

*in-kind contribution

Results from the joint SSM/Swedish NPP-funded project “Long term performance of polymeric materials in Nuclear power plants” will be taken into account in the work package.

2.1.1 Task 1 (T1.1) Ageing and functional testing

The goal of Task 1 is to develop and run the test method to be used when exposing the sample to ionized radiation and heat. This includes treatment in a chamber for low dose radiation, decision on what material properties to test with and what type of equipment’s to use. The test rig will need a suitable tube connector manufactured for the functional test. Depending on the polymer reaction products after irradiation and heat treatment can be used as well to study the changes in the material.

The test will run in a repetitive cycle with functional and property tests, exposure to radiation and exposure to heat. See figure 1 with the timeline showing where the tests are done and the time of each cycle. An estimation of
the acceptance criteria for when the function of the component is not satisfactory will be made and correlated to the compression set (if needed more material properties will be used).

2.1.2 Task 2 (T1.2) Implementation for the industry

The goal of Task 2 is to investigate the implementation of suitable methods in the daily operation at any of the nuclear power plants in the Nordic countries. The implementation phase will study the possibility for the plants to perform the needed tests on their own or if a separate lab needs to be consulted. Assistance in how a plant can set material properties for new components will also be discussed and presented. Methods developed can be used on verification of material quality of new components delivered by sub-contractor as well as verify material condition during annual services. This task will be performed in close collaboration with the interest group and aims to improve lifetime management of polymer. A telephone conference will be organised with the interest group to discuss the findings in the first ageing tests. Important topics are to understand how the plants can use the knowledge gained and what is missing. Can the results be used when purchasing new components and how can it be used to condition monitor existing components.

To test the possibilities for a power plant to use the knowledge gained from WP1 the Interest group will be asked to provide two real examples; one relating to the tested component, i.e an o-ring and one other example where monitoring during use could be done. The examples will be used to set out what type of tests that can be used, what material properties should be used for new components for instance.

2.1.3 Task 3 (T1.3) Modelling

The goal of Task 3 is to develop finite element (FE) models of the o-ring seals, which can be used to predict the leak tightness of different sizes and shapes of o-rings that have been exposed to heat and radiation. The model will use the compression set data acquired in Task 1 to tune the parameters of the material models used in the calculations. Additional testing of unaged material will provide the remainder of the needed parameters for the calculation model. Tests of the unaged material will consist of uniaxial tension and compression, and also a pure shear strain test. Validation simulations will be performed using the two different geometries that have been leak tested in Task 1. The calculation models will give the plants a broader knowledge about the functionality of the o-rings and tools which can be used to compare the performance of a wide range of components.

2.1.4 Proposed future work

There are four other identified components from the feasibility study that was concluded to be of interest for the development of acceptance criteria. These are Teflon seal, reinforced EPDM, joint sealants and lining materials. These components were highlighted during the interviews with the Nordic nuclear power plant representatives. However, it was decided to exclude these from the first work package and perhaps include them in a future follow up project, based on the results provided in WP1.

2.2 Work package 2 (WP2) Learn from materials used in plants – A feasibility study

The aim of this work package is to study materials undergone aging in operating and shut down power plants. This includes Barsebäck but also materials from outages in still operating power plants in the Nordic countries. This includes a pre-study to identify the polymeric components that can be available to study, analysis of the degradation of the selected materials and a workshop to present and discuss the results. A list of components will be compiled and a questionnaire to be sent to the Industry team for them to give valuable feedback on their interest and the possibilities to acquire materials from the plants.

It would be very valuable to compare artificially aged materials studied in WP1 with actual aged materials that have been in operation in a nuclear power plant. Many materials that are of interest can be obtained from the Nordic NPPs that are in operation, but this is not possible for some of the safety related polymeric components.
An alternative option could be to obtain materials and components from the closed down NPP Barsebäck or operating plants.

In this WP a pre-study will be performed in order to identify polymeric components that can be available from different plants in the Nordic countries, the amount of data that can be found on these components and information on if it is possible to gain access to these and possibly to take them off site.

The selection of components will be based on polymeric components included in WP1 to be able to correlate them to the accelerated tests. For the selection of materials, it will be important to consider for example if the type of material could be available at sites in operation today, and/or if the materials have been exposed to both high temperature and radiation. The latter may be significant to minimize the effect from storage after closure of the plant. It is also needed to take into consideration that since the reactor containment in a BWR is nitrogen filled during operation, i.e. that the time period after termination of power operation, might have exposed materials inside the containment to a much longer air period than during 30 years of operation.

The pre-study will also include finding the amount of data available on the components, and existing data for specific components will be used to show how a material has degraded. It is however likely that the possibility to find information will be limited. Chemical analysis will probably be needed to identify and characterize the materials further.

The access to materials will rather be limited to the radiological clearance than to availability of the material at the sites. Therefore, the pre-study will also include the evaluation of the process of taking the materials off from the site, and the option to perform any of the analysis on-site.

The identified components in the pre-study will be used to correlate the results from other proposed WPs with materials that have undergone ageing during operation for many years; this will yield valuable information on the ageing phenomena. They can also be used to investigate if the degradation is on the surface or in the bulk of the material. Presentation and discussion of the results is included in the work package.

Partners and person months allocated to WP2 to be given in the table.

<table>
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<tr>
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<th>Person months</th>
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*In-kind contribution

2.2.1 Task 1 (T2.1) Pre-study

The goal of task 1 is to fulfil the pre-study. Accessing Barsebäck materials were noted to be somewhat difficult due to the required radiological clearances and the documentation on the component service history seemed limited. These are the main reasons why this survey will be broadened to conclude also materials that would be available from currently running plants. The study would thus include: identification of the polymeric components that can be available to study, gathering the amount of data that exists on these today and the accessibility of these at the power plants. A “wishlist” that will contain all the materials that industry and project team are interested in to study is gathered and the materials and their properties on this list are later studied. The pre-study will also include a workshop to present and discuss the results.

2.3 Work package 3 (WP3) Polymer ageing mechanisms and effects inside NPP containments

The ageing environment for polymers within the containments of NPPs is rather complicated. During normal service of NPP the temperature within the containment can be tens of degrees beyond room temperature. Also radiation levels can vary from less than mGy/h up to ca. 1 Gy/h, depending on the reactor type. [Penttilä, 2016] So called hot spots with elevated temperature and higher dose rates are located in the vicinity of steam generator
tubing or in process valves. Thus polymer components are subjected to various stressors such as heat, radiation and moisture. In the both cases of thermal and radiation ageing, oxidation is considered to be the most common and dominating degradation mechanism. [Bartonicek & al] Oxidation of polymers is due to polymer radicals that are formed by absorption of thermal or radiation energy. These radicals react with oxygen forming peroxy radical which further reacts with the polymer chain forming hydroperoxide and polymer radicals. Hydroperoxide thermally decomposes to species that cause chain scission. Reaction mechanisms that govern radiation induced degradation of polymers have been more detailed described by [Schnabel 2014 and Makuuchi et al. 2011]. Under radiation, oxygen diffusion is known to be detrimental for polymers in room temperature and it is accelerated by increased temperature. Further complicating factor is DLO which has an effect to the heterogeneous oxidation behaviour of polymers. [Celina et al. 1996]

In Work Package 3 the effects of radiation and heat on polymer degradation are evaluated. It is known that most of the polymer degradation during normal usage of an NPP is due to thermo oxidative ageing but the effect of radiation ageing cannot be neglected. Especially during Design Based Accident (DBA) and severe accident scenarios the effect of radiation becomes more significant. A typical testing procedure for normal service conditions and polymeric materials includes separate irradiation and heat treatments. During irradiation conservative dose rates and accumulated doses are used. Such practice does not reflect the actual service situation because it does not take into account the synergistic effects of radiation and heat or the dose rate effect. To evaluate the testing methods for polymers used inside containments and acceptance criteria for these components, a fundamental understanding of the oxidation processes induced by radiation and thermal energies is needed or at least a quantitative estimation of their relative contribution to polymer degradation. To achieve such knowledge, literature surveys on the synergistic effects of radiation and heat as well as on methods that extrapolate dose rate effect from experimental data, were conducted during 2016. Based on the literature survey on synergistic effects of radiation and heat, the mechanisms governing the synergism are highly complicated. After the identification of key processes, one major complicating process related to polymer degradation was considered to be the reverse temperature effect. Research interest is focused on this phenomenon since it has a significant role on the degradation of semi-crystalline polymers and the phenomenon is not currently fully understood.

The synergy effects of radiation and heat were also determined experimentally on EPDM rubber and Lipalon cable jacket in 2016. The amount of degradation caused by thermo oxidative ageing, radiation ageing and their combined effect during conditions that are similar to a DBA were examined. Samples were exposed to thermal and radiation ageing separately and simultaneously and samples were analysed by tensile testing, hardness measurements and DSC and the dominating degradation effect was examined (thermal ageing vs. radiation ageing) and synergistic effects evaluated. The ultimate goal during 2017-2018 is to 1) propose improvements to accelerated ageing procedures to correspond better to realistic ageing environments i.e. normal service life and DBAs 2) Provide standard procedures for qualification which are detailed enough to improve the overall safety, but still easily understandable for both power companies and regulators. This would ease the qualification process and actual validation.

It is well known that oxidation of a polymeric material occurs at the material surface and the heterogeneous oxidation is affected both material thickness and temperature. The oxygen diffusion follows Ficks law and also the Arrhenius equation i.e the diffusion increases upon temperature increase. Degradation initiated by radiation may influence oxygen diffusion deeper into the bulk material and hence accelerate oxidation. Also the role of surface defects induced by surface oxidation in degradation of material properties needs to be clarified.

The dose rate effect has been recognized to be detrimental to some polymeric materials [Gillen 1981, Placek 2003] and as a part of this work package, the goal is to determine the significance of dose rate effect on polymers within the containment of NPP. The dose rate effect is related to the phenomenon where lower dose rates cause more degradation in the polymer properties than higher dose rates with the same total absorbed dose [Reynolds, IAEA-TECDOC551]. The diffusion of oxygen is closely related to this process since the diffusion of oxygen defines the depth of degradation within the polymer. With high dose rates all oxygen is consumed in the vicinity of the polymer surface and hence the damaged polymer structure is located near the surface and not in the bulk. With low dose rates oxygen has more time to diffuse in to the bulk and thus cause material degradation throughout the polymer.

As identified previously, dose rate effect exists at very high dose rates (DLO) as well as at very low dose rates. Dose rate effect exists on some polymers at very low dose rates even in inert atmospheres and thus the evaluation of significance of the effect is justified in order to validate the accelerated ageing treatments that simulate the
ageing at normal service conditions. Since the radiation levels during normal use in NPP are relative low (less than ~1 Gy/h) compared to the dose rates (10 kGy/h) defined in regulator instructors to be used during simulated ageing [Häkkä-Rönnhom, 2004], a predictive model that extrapolates the effects on lower dose rates would also be very beneficial. During 2016 these kinds of models, such as power law extrapolation method and superposition methods, were studied in a literature review. These methods require experimental data and it is produced as the dose rate effect is studied experimentally on EPDM and Lipalon cable jacketing materials.

Ageing management of NPP components is a well recognized field of nuclear safety by the international nuclear power community. Ageing management programmes related to polymeric components at NPPs aim to upkeep all the designed instrumentation and control systems that are required to function in appropriate and predetermined manner in order to maintain safe operation of the plant during normal service and DBAs [IAEA, 2000, 2012]. From the polymeric components the ageing management of cables has been identified to be an important safety factor, especially in case of NPPs that seek licence renewal after 40 years of service lifetime. [IAEA, 2012] However, literature or procedures related to ageing management of paint coatings, different sealants, lubricants and greases seem exist in lesser content in the open literature. As on part of this work package, a state-of-the-art survey on ageing management on polymeric materials (i.e. cables, sealants, paint coatings, lubricants and greases) used inside NPP containments is implemented within Nordic NPPs.

Partners and person months allocated to WP3 to be given in the table.

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<td>John Knott Associates Ltd.</td>
<td>0.05</td>
</tr>
</tbody>
</table>

*in-kind contribution

2.3.1 Task 1 (T3.1) Modelling tools for the synergistic effects of radiation and heat

In Task 3.1 atomistic simulation methods will be applied to study the reverse temperature effect, an anomalous ageing phenomenon observed in certain semi-crystalline polymers. The specific material to be considered is polyethylene, the degradation of which will be studied by means of classical and reactive molecular dynamics simulations. Separate model systems will be created for crystalline and amorphous polyethylene, as well as their interfacial regions. The key processes to be considered are: (i) radiation defect production (chain scission, crosslinking, formation of reaction intermediates) both with and without the presence of molecular oxygen and reaction intermediates, (ii) migration of molecular oxygen and the reaction intermediates, (iii) effect of elevated temperature on the previous. This task will contribute towards a mechanistic understanding of the reverse temperature effect. The research will be carried out during 2017 and 2018, with emphasis on the latter.

2.3.2 Task 2 (T3.2) Polymer ageing during service failure

During 2016 experimental data was used in determination on which ageing mechanism dominates during DBA (thermal vs. radiation) and in what kind of magnitude synergistic effects exist on EPDM and Lipalon cable jacketing. In 2017 no work is conducted in this subtask. In 2018 goal is to further evaluate how to set acceptance criteria for polymer components during normal service life and DBAs. Currently used qualification process (sequential ageing, no synergistic effects included) is evaluated. Based on results of obtain within this project and findings in STUK commission work [Penttilä et al. 2016] more sophisticated procedures for qualification and setting acceptance criteria are considered.
2.3.3 Task 3 (T3.3) Synergy effects between radiation and heat and oxidation depth

This task focuses on the surface oxidation in comparison to the bulk oxidation and the effect on overall material properties. EPDM samples exposed to different dose rates (described in Table 7) will be analysed in this task in order to evaluate the effect of dose rate to the formation of oxidation profile. The ToF SIMS analysis was considered to be a promising method for detection of oxygen at sample crosssection surfaces. From results of 2016 it was noted that more experience on sample preparation is needed in order to obtain smooth crosssections surfaces, which are required for proper analysis results. This technique will be used to detect differences in the oxidation behaviour of EPDM induced by different dose rates. An additional SEM evaluation of samples is conducted in order to visualize the microscale degradation and evaluate the microstructural changes in crystalline and amorphous regions of the samples.

2.3.4 Task 4 (T3.4) Evaluation of damage caused by dose rate effect to polymer components used within containments

The goal of this task is to provide information about the dose rate effect on polymer materials used inside NPP containments which can then be used in the improvement of ageing management of these components. One important aspect is to evaluate whether dose rate effect is so significant that it should be taken into account during the qualification of polymer materials. During year 2016 different methods that can be used in extrapolation of dose rate effect were evaluated and during 2017 experimental data is gathered at relative high dose rates (compared to normal service conditions of NPP) and extrapolation is conducted according to suitable methods during late 2017 and 2018. As a result, estimation about severity of dose rate effect can be conducted for the studied materials.

The dose rate effect is studied for two materials which are used at NPPs: EPDM and Lipalon cable jacketing material. They are irradiated according to the irradiation treatment program described in Table 7. Most of the samples are irradiated at UV Rez, Czech Republic and in order to save in costs single irradiations with long treatment times (marked with * in Table 7) are conducted with VTT’s own gammacell. The actual dose rates are confirmed after dose mapping and values presented in Table 7 are the current best estimates available. Three times five EPDM dumbbells and three times five Lipalon cable samples as well as three times one EDPM sheet for DLO studies are included in each treatment, since each treatment will include three different dose rates. The dose rates included in treatment 1 are at similar magnitude as the dose rates used in current accelerated irradiation ageing treatments (e.g. such as described in [Häkkä-Rönnholm, 2004]) and it provides data on DLO effects on material degradation as the dose rate is slightly decreased below the required standardized limit of 10 kGy/h. Treatment 2 will have similar total absorbed dose that is cumulated during a DBA. The effect of dose rate is again studied in order to evaluate whether a ten time decrease in dose rate could produce any dose rate effects on the studied sample materials. In the case of treatment 3, the total dose corresponds the cumulative dose after 40 years of normal service (dose rate of ca. 0.114 Gy/h). Again, any signs of dose rate effect becoming apparent are evaluated by using five different dose rates. Treatments 4 and 5 can be included in the radiation procedure without any extra costs due to the relatively low absorbed dose and they provide additional data for the semi-empirical superpositioning models that can be used in dose rate effect extrapolating. And if available, additional EPDM samples that are removed during outages are tested here for compare the two EDPM qualities. EPDM and Lipalon cable jacketing samples are tensile tested and their hardness is measured. DLO measurements are conducted in task 3.3.

The dose rate effect was noted that more experience on sample preparation is needed in order to obtain smooth crosssection surfaces. This technique will be used to detect differences in the oxidation behaviour of EPDM induced by different dose rates. An additional SEM evaluation of samples is conducted in order to visualize the microscale degradation and evaluate the microstructural changes in crystalline and amorphous regions of the samples.

Table 7. Irradiation treatments in task 3.4. More exact dose rates will be confirmed after dose mapping. Irradiation treatments marked with * are conducted with the gammacell located at VTT.

<table>
<thead>
<tr>
<th>Treatment number</th>
<th>Total dose (kGy)</th>
<th>Dose rate (kGy/h)</th>
<th>Dose rate (kGy/h)</th>
<th>Dose rate (kGy/h)</th>
<th>Dose rate (kGy/h)</th>
<th>Dose rate (kGy/h)</th>
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</thead>
<tbody>
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<td>1</td>
<td>1000</td>
<td>6</td>
<td>3</td>
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<td>N/A</td>
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<td>2</td>
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<td>0,06*</td>
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<td>3</td>
<td>40</td>
<td>6</td>
<td>2</td>
<td>0,6</td>
<td>0,36</td>
<td>0,06*</td>
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<tr>
<td>4</td>
<td>20</td>
<td>6</td>
<td>2</td>
<td>0,6</td>
<td>N/A</td>
<td>0,06*</td>
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<tr>
<td>5</td>
<td>2</td>
<td>6</td>
<td>2</td>
<td>0,6</td>
<td>N/A</td>
<td>0,06*</td>
</tr>
</tbody>
</table>
2.3.5 Task 5 (T3.5) State-of-the-art of ageing management of cables, sealants, paint coatings, lubricants/greases used inside Nordic NPP containments

Ageing management programmes can have variations between NPPs and knowledge related to the fact that these programmes are managed can be scattered within the NPP and regulator personnel. The first step would be to gather the right people from the Nordic authorities and NPP personnel that have the current best knowledge on ageing management on cables, sealants, paint coatings, lubricants and greases. Once the right people have been contacted, a series of interviews are conducted with the ageing management experts. During the interviews information on the following points are collected:

1) Current understanding on the degradation mechanism causing the ageing on different components
2) Effect of ageing to component functionality
3) The use of predictive models on components or evaluation methods for component lifetime
4) Procedures for surveillance and detection of ageing induced degradation
5) Material suppliers’ ability to deliver good enough quality components
6) Procedures for detecting faulty components before their installation
7) Procedures for maintenance and repairs
8) Documentation of ageing issues and related knowledge bases

Based on the results from the interviews and the search through relevant literature, a state-of-the-art report is prepared. Within the report above points are evaluated and the use of scientific ageing research to provide technical basis and supporting ageing management at plants is considered. Also boundary conditions for component specific ageing management programs are discussed and currently existing standards related ageing management of these components gathered.

There is no work planned in this task during 2017.

2.4 Work Package 4 (WP4) International cooperation

Within this WP international cooperation is promoted. This is done by organizing meetings and educational events with nuclear power industry members, researchers and regulators. Aim is to gain synergistic advantages from large Nordic consortium in research activities, nuclear safety issues and improved plant efficiency.

Table 8: Partners in WP4.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
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<tbody>
<tr>
<td>VTT</td>
<td>0.25</td>
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<tr>
<td>John Knott Associates Ltd.</td>
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Task 1 (T4.1) COMRADE workshop/seminar

A two-day meeting is organized by VTT between the project team, Nordic industry, regulator and polymer manufacturer representatives in September 2017. The meeting will include a tailored lecture on degradation of polymers inside NPPs given by Dr. Sue Burnay (represents John Knott Associates Ltd.), presentation and discussion on project results and excursion on VTT laboratory premises. Similar meeting was organized by SP and Energiforsk in 2016 in Borås, Sweden and it was considered to be successful. These meetings will strengthen the cooperation between the participants and provide new contacts and knowledge related to polymer degradation issues within NPPs.
### 3. Deliverables 2017

Table 9. List of all deliverables planned for the project year 2017.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1</td>
<td>Development of condition monitoring technique for O-rings used in NPP applications (mid-term presentation at the COMRADE workshop)</td>
<td>2,1(^1)+0,5(^2)</td>
<td>09/2017</td>
</tr>
<tr>
<td>D1.2</td>
<td>Development of FEM model and test run using data from T1.1 (SP-research report)</td>
<td>1,4(^1)</td>
<td>09/2017</td>
</tr>
<tr>
<td>D1.3</td>
<td>Organize a telephone conference with the industry team to present and discuss how the result form the first ageing test can be used.</td>
<td>0,3</td>
<td>06/2017</td>
</tr>
<tr>
<td>D2.1</td>
<td>Identification of available polymers and their data nuclear power plants both running and shut down. A small scale test is also included to verify towards model in T1.3 (including one workshop and SP-research report)</td>
<td>0,4(^4)+0,05(^2)</td>
<td>11/2017</td>
</tr>
<tr>
<td>D3.1</td>
<td>Modelling efforts to understand the reverse temperature effect (mid-term presentation at the COMRADE Workshop)</td>
<td>1,0(^2)</td>
<td>09/2017</td>
</tr>
<tr>
<td>D3.2</td>
<td>Dose rate effect study on EPDM and Lipalon cable jacketing materials (VTT Research Report)</td>
<td>0,4(^1)+1,8(^2)</td>
<td>12/2017</td>
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<tr>
<td>D4.1</td>
<td>COMRADE Workshop/seminar on polymer ageing issues at NPPs (Minutes of the meeting)</td>
<td>0,25(^2)</td>
<td>09/2017</td>
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<td><strong>Total pm</strong></td>
<td><strong>8,2</strong></td>
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</table>

\(^1\)SP person months
\(^2\)VTT person months
4. Project organisation

The project is implemented as cooperation of VTT and SP. Konsta Sipilä (VTT) will act as the responsible person from VTT and for the overall project while Marcus Granlund (SP) will act as the responsible person from SP. Together with the team from VTT and SP there will be an interest group from the NPPs and other interested parties. The group will help to ensure industry/authority relevance as they are invited to take an active part in the project.

The project will report to a reference group appointed by the SAFIR2018 Management board.

Table 10. The project organisation at VTT and SP.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
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<tr>
<td>Anna Jansson</td>
<td>Senior scientist</td>
<td>SP</td>
<td>T1.1, T3.3</td>
<td>0,8</td>
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<tr>
<td>Anna Bondesson</td>
<td>Research scientist</td>
<td>SP</td>
<td>T2.1</td>
<td>0,4</td>
</tr>
<tr>
<td>Marcus Granlund</td>
<td>Research scientist</td>
<td>SP (Responsible SP)</td>
<td>T1.1, T1.3, T2.1, T3</td>
<td>2,0</td>
</tr>
<tr>
<td>Johan Sandström</td>
<td>Research scientist</td>
<td>SP</td>
<td>T1.3</td>
<td>1,4</td>
</tr>
<tr>
<td>Jan Henrik Sällström</td>
<td>Research scientist</td>
<td>SP</td>
<td>T1.3</td>
<td>0</td>
</tr>
<tr>
<td>Harri Joki</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T3.4</td>
<td>1,15</td>
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<td>Antti Paajanen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T3.1</td>
<td>0,4</td>
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<tr>
<td>Tiina Lavonen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>0,9</td>
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<tr>
<td>Jukka Vaari</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T3.1</td>
<td>0,5</td>
</tr>
<tr>
<td>Konsta Sipilä</td>
<td>Research scientist</td>
<td>VTT (Responsible VTT and overall project)</td>
<td>T1, T2, T3, T4</td>
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<tr>
<td><strong>Total</strong></td>
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<td></td>
<td></td>
<td><strong>8.2</strong></td>
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</table>

An interest group will work with the project and be invited to participate during the planned workshops. The group consists of staff from different parts of the industry, see table below.

Table 11. The Interest group.

<table>
<thead>
<tr>
<th>Name</th>
<th>Organisation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monika Adsten</td>
<td>Energiforsk AB</td>
</tr>
<tr>
<td>Eric Jansson</td>
<td>EON, OKG</td>
</tr>
<tr>
<td>Lauri Rintala</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Juha Rinta-Seppälä</td>
<td>Fennovoima</td>
</tr>
<tr>
<td>Ritva Korhonen</td>
<td>Fortum</td>
</tr>
<tr>
<td>Kristiina Söderholm</td>
<td>Fortum</td>
</tr>
<tr>
<td>Jukka Sovijärvi</td>
<td>STUK</td>
</tr>
<tr>
<td>Pekka Vallikangas</td>
<td>STUK</td>
</tr>
<tr>
<td>Liisa Heikinheimo</td>
<td>TVO</td>
</tr>
<tr>
<td>Timo Kukkola</td>
<td>TVO</td>
</tr>
<tr>
<td>Henrik Widstrand</td>
<td>Vattenfall</td>
</tr>
<tr>
<td>Annelie Jansson</td>
<td>Vattenfall, Forsmark</td>
</tr>
<tr>
<td>Stjepan Jagunic</td>
<td>Vattenfall, Ringhals</td>
</tr>
<tr>
<td>TBD</td>
<td>Barsebäck</td>
</tr>
<tr>
<td>John Rogers</td>
<td>James Walker</td>
</tr>
</tbody>
</table>
5. Risk management

Table 12. Risk management plan. To be updated during the project on a periodic basis.

<table>
<thead>
<tr>
<th>Risk</th>
<th>Probability of occurrence</th>
<th>Potential impact on project success</th>
<th>Mitigation Plan</th>
</tr>
</thead>
<tbody>
<tr>
<td>Significant changes in the research plan.</td>
<td>Low</td>
<td>Medium</td>
<td>- Study the impact of the changes on schedules and results</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Implement changes, if the impact is high</td>
</tr>
<tr>
<td>Poor data quality</td>
<td>Low</td>
<td>High</td>
<td>- Only use records which have good quality basic data sets</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Only use qualified staff for testing</td>
</tr>
<tr>
<td>Costs could rise significantly during the time of the project</td>
<td>Low</td>
<td>High</td>
<td>- Monitor costs on a periodic basis</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Using other funds.</td>
</tr>
<tr>
<td>Loss of key researcher (unable to complete key tasks)</td>
<td>Low</td>
<td>High</td>
<td>- Identify alternative resources in case of unexpected absence.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Ensure complete records of work are available at any point</td>
</tr>
<tr>
<td>New cooperation between SP and VTT</td>
<td>Low</td>
<td>Medium</td>
<td>- Arrange a project start up meeting in the beginning of the project.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Include periodic project meetings to follow progress and cost</td>
</tr>
</tbody>
</table>
References


IAEA. Assessement and management of ageing of major nuclear power plant components important to safety: In-containment instrumentation and control cables, TEC doc 1188. Vienna, Austria: IAEA. (2000)


Widestrand, 2015. Email provided to M. Granlund sent from H Widestrand 2015-10-19.

### Work packages and Tasks

<table>
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<tr>
<th>Expenses</th>
<th>Financing</th>
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<td>Work packages and Tasks</td>
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**Comments:**

The person-months shown in the table (8.2) is the work conducted by VTT and SP personnel. In addition there will be in-kind contributions from Nordic NPPs 0.6 PMs and James Walker 0.2 PMs.

Other explanatory comments:

In WP1 Task 1.1 there is other cost (11.0 ke) which is cost for running testing equipment at SP.

In WP1 (1.4 ke), WP2, WP3 and WP4 the cost of VTT research facilities are shown in "Others" section.
## VTT share

### Expenses

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<th>Work packages and Tasks</th>
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<th>Personnel</th>
<th>Mat&amp;supp</th>
<th>Travel</th>
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SAFIR2018 Project plan for 2017 application

ERNEST

Experimental and numerical methods for external event assessment improving safety

Ari Vepsä
Senior Scientist
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1. Research theme and motivation

An aircraft impact on safety related structures, in spite of its low probability, has for a long time been recognized as a relevant loading case, especially in designing plants to areas with heavy air traffic. It is required in Government Decree 733/2008 (1) that the nuclear plant design takes into account large airliner crashes, as well as resulting fires and explosions (2), (3). Structural analyses of these phenomena require nonlinear numerical analysis methods. In order for the results of these numerical analyses to be reliable, the applicability of the used methods and models should be validated by experimental results and analytical methods.

The numerical methods developed in this research can be applied for safety and structural integrity assessments of both operating NPPs and those to be built in the future. For existing plants, structural safety margin assessment is of primary interest. On the other hand for new-built NPPs, these methods can be applied in design and optimisation of protective structures and in particular of sacrificial structures. In addition to aircraft impacts, these methods can be effectively used in the analysis of structures under dynamic loads that are caused by external and internal explosions (e.g. severe accidents).

Computational methods and tools are under continuous development and the computation capacity is exponentially increasing. Thus, more and more realistic numerical simulations can be carried out with reasonable efforts. Different types of numerical methods can now be coupled to solve multi-physical problems. Modelling of a heavy projectile impact on a NPP gives valuable beyond code design information that enables predictions concerning mechanical responses of internal components and instruments. Methods for numerical simulation of induced vibrations in damaged concrete need to be developed further and taken in use.

Experimental work is the cornerstone of this study, because it is needed in the understanding these highly complex phenomena. In order to make a successful test careful planning and pre-analysis is required. Post-analysis is done after the test results are available using the same model with actual parameters obtained from the test results. By comparing series of tests with corresponding numerical analysis results, one can determine the applicability of the used model and method to describe the phenomenon under consideration. Further, conclusions for development needs can be drawn. Figure 1 shows the schematics of this approach.

![Figure 1. Development of computational tools utilizing experimental data.](image)
1.1 Background and state-of-the-art

The international IMPACT project managed by VTT has produced valuable test data on impact loaded reinforced concrete (RC) structures since 2003. Successful tests require good planning and relevant pre-analyses. Methods have been developed and the applicability of the already existing methods to different cases has been tested in earlier predecessor projects. The predecessor projects of these studies were carried out within Research area 7 of SAFIR2014 program. In the evaluation report (4) it was stated that “the work is world leading and involve impressive combination of experimental and analytical expertise to address important topics in the subject areas and provide a basis for very valuable international collaboration”. The recommendation was that the present high level work should be carried on by enhancing international collaboration. Resources are needed for test planning with international partners, to participate in workshops and to cooperate in the IMPACT 4 project.

The missile of the real design case, an aircraft, is a combination of deformable and hard parts. These types of missiles are tested separately in the IMPACT project. Failure mode due to an impact of a hard missile is mainly local penetration or perforation. The local wall damage can be predicted at simplest by semi-empirical formulas or in more detail by finite element (FE) models. The traditional and widely used method to model an impact by a deformable missile is to apply the Riera formula. In this formulation the impact load is determined based on the crushing force and mass flow of the missile. According to the previous studies, force-time function calculated by the Riera formula has been proved to be reasonably accurate for solely target deflection behaviour. However, it should be kept in mind that the impacts of deformable and hard parts, such as engines and fuselage, occur simultaneously. Simulations that reproduce this complex behaviour accurately can be done using coupled models where both the target and the missile are properly described. Recently, calculation results on coupled analyses have been published with the aircraft modelled explicitly (5), (6). This approach may affect especially the vibration behaviour of structures, e.g. buildings housing safety related equipment.

No perfect method seems to exist for nonlinear dynamic analyses of reinforced concrete structures and thus different kinds of approaches are needed. According to the previous studies, a nonlinear finite element (FE) model utilising explicit time integration and a simple four-node shell element without considering transverse shear deformation is capable of predicting the deflection behaviour of a reinforced concrete wall loaded by a deformable missile. In case the wall collapses in bending mode, the maximum deflection can be predicted reasonably well. Bending and punching can be modelled by shell elements where the nonlinear transverse shear deformation is included. In geometrically detailed studies and in case of perforation three dimensional solid models with a relevant element erosion technique are needed. Applicability and conservatism of different methods need to be evaluated and assessed.

Although extensive nonlinear models are often needed in analysing impact loaded reinforced concrete structures, much simpler models can be utilized for preliminary safety assessment. Simplified methods, such as semi-empirical equations and models comprising of only few most essential degrees-of-freedom, can especially well be applied for parametrical studies and preliminary design, as shown in (7), (8) and (9).

Both aircraft impacts and earthquakes induce vibrations that propagate throughout the entire building. These vibrations need to be considered in designing SSCs (Structures, Systems and Components). One of the main challenges in the field of nuclear power is the increased use of digital equipment that is sensitive to potential environmental stressors such as vibrations. So far, mainly linear calculation methods have been used and thus the codes and standards consider only damping ratios for linear assessments. Induced vibrations, especially in damaged concrete structures, have not been studied extensively enough. Especially experimental data on damping properties of damaged reinforced concrete is needed. Recently, within the IMPACT project a new type of test series has been carried out. During this test series, propagating vibrations generated by impact loads and the role of damping are studied. The gathered data provides a good basis for the development and validation of numerical methods and models. The vibration phenomenon is also the main objective in the next OECD/IAGE/IRIS3 benchmark exercise. This IRIS test was conducted at VTT in October 2016.
1.2 Objectives and expected results

The main aim of this work is to develop and take in use improved methods and modelling techniques which are validated against experimental results. Experimental data on nonlinear dynamic behaviour of reinforced concrete structures loaded by hard and deformable missiles has been obtained at VTT within IMPACT projects (Phases 1, 2 and 3) already for over 10 years. The next project, IMPACT 4, is now under planning. International partners from previous projects are willing to continue the cooperation. The pre- and post-studies related to the IMPACT project will be carried out within the ERNEST project. Some dedicated tailored tests for domestic purposes will be undertaken within the ERNEST project.

Models and methods for assessing structural integrity of impact loaded reinforced concrete structures are developed and validated utilising experimental data. In practice, post calculations of impact tests are an important way to identify needed development work. Knowledge transfer, training and education of new experts is carried out within this kind of working process. The results will be reported mainly as scientific and conference papers.

1.3 Exploitation of the results

Methods and modelling techniques developed and validated here can directly be applied in safety assessment and design analyses. Results obtained within these studies are useful for safety authorities and utilities in structural safety assessment of both operating and new built NPPs. These methods can be applied in structural safety margin assessment, design and optimisation of protective structures. Development work of sacrificial structures is of paramount interest for end users willing to protect the plants with maximal efficiency.

1.4 Appropriateness of the project to SAFIR2018 programme

The importance and need for further development of advanced assessment methods for structural integrity studies is stated in Chapter 3.4.4.4 of SAFIR2018 framework plan. It is also pointed out that all the relevant test results should be used in the development work. This primarily concerns the utilization of experimental data gathered within the ongoing IMPACT 3 and the forthcoming IMPACT 4 projects. The developed material model can be utilised also when assessing ageing of concrete structures.

International and national cooperation

The following channels for cooperation will be exploited during the project:

- Workshops on numerical analyses will be organized jointly within the Technical Advisory Group meetings of the international IMPACT project
- Participation in ERNCIP (EUROPEAN REFERENCE NETWORK FOR CRITICAL INFRASTRUCTURE PROTECTION) thematic group ‘Resistance of structures to explosion effects’, coordinated by EMI (Ernst Mach Institute)
- Participation in 1st International conference on Nuclear Power Plants NUPP 2017 (abstract accepted).
- Cooperation with Aalto University and Tampere University of Technology

1.5 Education of experts

One new research scientist has been recruited and the training is going on. Another research scientist will be trained for numerical simulation work in order to enforce the available resources. Knowledge transfer from prof. emeritus M. Tuomala to VTT experts will be carried out within this project. Joint papers will be prepared to scientific journals and conferences.
2. Work plan

2.1 Work package 1 (WP1) - Experimental studies

The general goal of the WP is to produce experimental data about different impact-induced phenomena on different structure types to satisfy needs of computational model validation and development and also for more general knowledge of the subject. Especially, tests for domestic needs will be conducted here.

During the forthcoming project years 2017 and 2018, both static tests and dynamic impact tests have been planned to be carried out. The details of the tests have not been fixed yet, especially for 2018. In 2015, a journal paper was written regarding vibration propagation and damping test series, which was carried out in an earlier testing project. In addition, the measurement data acquisition and processing software was updated with few new features which makes it easier to be used by the measurement personnel. In 2016, two impact tests have been carried out towards the end of the project year.

2.1.1 Task 1.1 - Punching behaviour test with two consecutive slabs

In this task, a combination of two consecutive reinforced concrete slabs separated with a gap will be tested by impacting a hard projectile against the foremost slab. Besides aircraft impact generated missiles, also other types of missiles e.g. missiles generated by turbine failure can be considered. The idea of the proposed twin-slab test is to study how well this type of structural system can resist perforation by a hard projectile when it is impacted against such a slab combination. This type of structural solution is one possibility to safeguard nuclear power plant buildings against the harmful effects of an airplane crash. The purpose of the first slab is to act as a sacrificial structure which dissipates as much of the impact energy as possible so that damages caused for the posterior slab are diminished. In addition to perforation resistance, the amount of scabbing on the distal surface of the posterior slab is another important output parameter. The reference case is a single slab with the same amount of reinforcement and the same concrete properties. Comparison between tests would give information regarding feasibility of these two structural choices.

Previously, 250 mm thick simply supported reinforced concrete slabs with span width of 2.0 metres in both directions have been tested at VTT for hard impacts. These tests are used as reference cases against which the results from the proposed test are compared. Therefore the proposed test is to be carried out with a two 125 mm thick slabs with a gap between them. The impacting projectile should be as rigid as possible with the weight being 47.5 kg. The nominal strength of concrete used for the slabs shall be K50/C40 with the maximum grain size being 8 mm to comply with the comparison tests. The main output parameters are the exit velocity of the projectile if it perforates both slabs, the penetration depth in case it does not and the scabbing area of the rear surface of the posterior slab. The test would be carried out with VTT’s impact testing apparatus during the early autumn of 2017 and reported in a technical research report.

2.2 Work package 2 (WP2) - Modelling of nonlinear behaviour of RC structures

The main aim of this work is to develop and take in use improved methods and modelling techniques for dynamically loaded reinforced concrete structures. Analysis methods are validated against experimental results. A lot of experimental data on nonlinear dynamic behaviour of reinforced concrete structures loaded by different types of hard and soft (deformable) missiles has been created at VTT within IMPACT projects (Phase 1, 2 and 3)
already for over 10 years. In 2015, within NEST project, studies on vibration propagation were carried out along
the available test results and a scientific journal manuscript was prepared in cooperation with the IMPACT project
partners. Three conference papers were prepared and presented in the SMiRT 23 conference. A joint report with
Scanscot (Sweden) was prepared on VERCORS benchmark studies. In 2016, within ERNEST project, develop-
ment of a new concrete material model was started. Three abstracts were submitted to the SMiRT 24 conference.

Development, validation and qualification of calculation models and methods are carried out by utilising availa-
ble impact test data. Pre-studies for IMPACT 4 tests will be conducted when the tests are planned and post calcu-
lations on selected impact tests will be continued. The main aim is to test and validate existing methods and mod-
elling techniques. Also possible development needs will be identified and actions taken within the budget frames.
In 2018, the experimental data obtained from the ongoing water filled missile test series (IMPACT 3) will also be
utilised in developing calculation tools.

Verified methods for modelling damping in nonlinear reinforced concrete are incomplete, e.g. the effect of
damage degree on damping properties of concrete in cyclic loads needs to be studied in more detail. Experi-
mental observations and measurement data will be utilised in developing calculation tools for vibration propaga-
tion analyses.

2.2.1 Task 2.1 - Concrete material model development

The development of dynamic simulation methods to simulate reinforced concrete structures under impact load-
ing is a core issue in this project. Recent work (5) has shown that commercially available concrete models, such
as the Concrete Damage Plasticity (CDP) model proposed in Abaqus, (10), have trouble dealing correctly with
extreme load case scenarios involving massive fracturing and material disaggregation due to projectile penetra-
tion or seismic loading. Resources were spent to upgrade the Abaqus CDP model to deal with ultimate failure
analysis, which means that the first assignment shown in table below is already well under progress.

The aim of Task 2.1 is to upgrade the currently used commercially available material models to scientific state-
of-the-art models documented in the literature with the help of user defined material subroutines in Abaqus. The
approach to develop a new material model has to be methodical and iterative, involving verifications against re-
ported test data at the end of each assignment. The assignment can be summarized in the following table:

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<td>Projectile penetration simulations. Progressive collapse in seismic and impact simulations.</td>
<td>Computation of internal dissipation variable and cohesion stress on output including strain rate dependency. Implementation of element deletion criteria depending on the current triaxial stress/strain state and the cohesion level.</td>
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The assignment CDP model for ultimate failure analysis can be implemented, at first, using the standard CDP
model provided by Abaqus and by enhancing it with user subroutines. In the end it would be tempting, however,
to program a stand-alone Abaqus user material in order to get a better insight on the internal behaviour.

Testing and calibration of the material models involves simulations of single elements to calibrate the local re-
sponse as well as simulations on benchmark structures. Relevant material test data for the uniaxial cube com-
pression test, the Brazilian tension-split test, the split-Hopkinson pressure bar test in tension and compression
can be obtained in the open literature. Experimental data for validation purposes is obtained from IMPACT test
results (phases 1-3). Developed model will be further applied in Tasks 2.2 and 2.3.

2.2.2 Task 2.2 - Pre- and post-analyses for IMPACT and ERNEST tests

As mentioned above, the international IMPACT project managed by VTT has produced valuable test data on
impact loaded reinforced concrete structures since 2003. Methods have been developed and the applicability of
the already existing methods to different cases has already been tested in earlier predecessor projects. However,
the testing is a continuous effort and the methods and calculation models have to be validated, qualified and de-
Impact and material testing will be continued in the next phase of IMPACT project, but also in Task 1.1 of this ERNEST project.

Pre-analyses are needed for the planning of the tests in order to be able to choose proper test parameters for ensuring safety and for ensuring that the desired physical phenomena are present and as much information from the tests as possible can be obtained. Planning of IMPACT tests will be carried out in co-operation with the international partners. Post-analyses are needed for validation of the computational methods and identification of issues to be improved.

Different kinds of modelling approaches are needed. Existing approaches are suitable either for hard or soft projectile impacts. A long term aim is to develop a unified approach capable of dealing with all types of impacts. However, all scenarios always need to be studied with more than one approach for verification of the analysis results.

2.2.3 Task 2.3 - Vibration propagation and damping

Methods for modelling vibration propagation in nonlinear reinforced concrete are incomplete. The effect of concrete cracking on damping properties needs to be studied in more detail.

Vibration propagation test series V0A-C, V1A-F and IRIS phase 3 tests provide valuable information for better and more comprehensive understanding of this a rather complicated and important phenomenon. Also the experimental results from bending wall tests carried out during IMPACT 1-3 projects are utilised in studying the vibration behaviour of a damaged concrete wall.

Studies on selected tests will be continued in 2017. This work will concentrate on realistic numerical simulation of vibration propagation in damaged reinforced concrete. Material model developed carried out in Task 2.1 can also be used here.
## 3. Deliverables 2017

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<th>Indicative person months</th>
<th>Deadline date</th>
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<tr>
<td>N/A</td>
<td>The proposed impact test carried out</td>
<td>N/A</td>
<td>15.9.2017</td>
</tr>
<tr>
<td>D1.1.1</td>
<td>Research report discussing the proposed test (Tasks 1.1, 2.2)</td>
<td>1.5</td>
<td>31.10.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Conference paper on material model development (Task 2.1)</td>
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<td>31.5.2017</td>
</tr>
<tr>
<td>D2.2.2</td>
<td>Research report on pre and post analyses (Task 2.2)</td>
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<tr>
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<td>Conference papers on damping studies (Task 2.3)</td>
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<tr>
<td><strong>Total pm</strong></td>
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4. Project organisation

Project manager is M.Sc. (Tech.) Ari Vepsä and deputy project manager is Lic.Sc. (Tech.) Arja Saarenheimo. VTT is responsible for the whole project.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ari Vepsä</td>
<td>Project manager</td>
<td>VTT BA2505</td>
<td>T1.1</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>Senior Scientist</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Arja Saarenheimo</td>
<td>Research Team Leader</td>
<td>VTT BA2505</td>
<td>T2.3</td>
<td>1.5</td>
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<tr>
<td>Kim Calonius</td>
<td>Senior Scientist</td>
<td>VTT BA2505</td>
<td>T2.2</td>
<td>1.5</td>
</tr>
<tr>
<td>Alexis Fedoroff</td>
<td>Research Scientist</td>
<td>VTT BA2505</td>
<td>T2.1</td>
<td>1.5</td>
</tr>
<tr>
<td>Markku Tuomala</td>
<td>Prof. Emeritus</td>
<td>-</td>
<td>T2.2, T2.3</td>
<td>1</td>
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<tr>
<td>Jukka Mäkinen</td>
<td>Senior Research Technician</td>
<td>VTT BA2401</td>
<td>T1.1</td>
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<tr>
<td>Mikko Kallio</td>
<td>Research Engineer</td>
<td>VTT BA2402</td>
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<td></td>
<td></td>
<td></td>
<td><strong>7</strong></td>
</tr>
</tbody>
</table>
5. Risk management

One general risk concerning the whole project is personnel resources, especially the availability of key persons. Also loss of young experts coming to the project with previous knowledge build-up in a few years, or in advanced stage of training within the project, would be very difficult to compensate from the project targets point of view. Hence, it is important to offer professional incentives within the project to carry on focusing on these fields of study.

Besides, the following punctual risks have been identified and corrective actions planned:

<table>
<thead>
<tr>
<th>Task</th>
<th>Risk</th>
<th>Likelihood estimate (%)</th>
<th>Check-it date</th>
<th>Fall-back plan</th>
<th>Action / Monitoring activity</th>
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<tbody>
<tr>
<td>1.1</td>
<td>The test fails for some reason.</td>
<td>&lt;0.5%</td>
<td>1.10.2017</td>
<td>None. If the test fails, it fails and it cannot be taken back or corrected. The reason for the failure will be sought and the lesson from it will be learned so that it wouldn't happen again.</td>
<td>RG2 and the ad-hoc group will be informed on the situation.</td>
</tr>
<tr>
<td>2.1</td>
<td>Drawbacks inherent to research work.</td>
<td>20%</td>
<td>1.11.2017</td>
<td>Work will be continued during the next year and cooperation will be increased.</td>
<td></td>
</tr>
<tr>
<td>2.2</td>
<td>Tests in Task 1.1 are delayed or have failed.</td>
<td>10%</td>
<td>1.10.2017</td>
<td>In a case of delay: post-analyses will be postponed and a substitute case study will be selected. In case of failure: a new improved test plan for 2018 will be made and pre- and post-calculations will be carried out accordingly.</td>
<td>RG2 and the ad-hoc group will be informed on the situation. Substitute case studies will be discussed.</td>
</tr>
<tr>
<td>2.2</td>
<td>The international IMPACT 4 project is delayed</td>
<td>20%</td>
<td>1.2.2018</td>
<td>In a case of delay: Focus on the existing test results</td>
<td>RG2 and the ad-hoc group will be informed on the situation. Substitute case studies will be discussed.</td>
</tr>
</tbody>
</table>
References


## ERNEST

Experimental and numerical methods for external event assessment improving safety

### Work packages and Tasks

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
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<td>Volume</td>
<td>Personnel</td>
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<tr>
<td>WP1 - Experimental studies</td>
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<td>Task 1.1 Punching behaviour test with two consecutive slabs</td>
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<tr>
<td>WP2 - Modelling of nonlinear behaviour of RC structures</td>
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<td>60</td>
</tr>
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<td>Task 2.1 Concrete material model development</td>
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<td>19.875</td>
</tr>
<tr>
<td>Task 2.2 Pre- and post-analyses for IMPACT and ERNEST tests</td>
<td>1.5</td>
<td>19.875</td>
</tr>
<tr>
<td>Task 2.3 Vibration propagation and damping</td>
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<td>19.875</td>
</tr>
<tr>
<td>TOTAL</td>
<td>6</td>
<td>79</td>
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</table>

**Comments:** Computation platform cost in Other costs
SAFIR2018 Project plan

FIRED

Fire Risk Evaluation and Defence-in-Depth

Anna Matala
Research Scientist, VTT

Simo Hostikka
Associate Professor, Aalto University

Antti Paajanen
Research Scientist, VTT

Topi Sikanen
Research Scientist, VTT
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1. Research theme and motivation

A significant proportion of the overall core damage risk in nuclear power plants (NPP) is associated with internal fires. In addition, a fire on NPP can cause large financial losses even if the risk to the reactor safety was small. Therefore the possible initiating event scenarios and the operation of defence-in-depth after ignition are important topics in the research of nuclear safety. The computational tools that are used for assessing the fire risks have developed significantly over the last ten years: The deterministic analyses are now solely based on CFD and the probabilistic analyses using Monte Carlo simulation have been carried out. These developments have improved the reliability and accuracy of the safety analyses. In addition, the same deterministic tools are also used for the analyses of external fire incidents, such as aircraft -impact induced fires. Maintaining the computational infrastructure (tools, hardware and competence) requires continuous investments, and significant research efforts are still needed to enable predictive engineering simulations of fire spreading.

FIRED-project will cover three main themes: fire risks of cables during the plant life cycle, fire Defence-In-Depth, and modelling tool development and validation. In addition, active participation to OECD PRISME 2 – project will continue.

1.1 Background and state-of-the-art

A significant proportion of the overall core damage risk in nuclear power plants (NPP) is associated with internal fires. In addition, a fire on NPP can cause large financial losses even if the risk to the reactor safety was small. Therefore the possible initiating event scenarios and the operation of defence-in-depth after ignition are important topics in the research of nuclear safety. The computational tools that are used for assessing the fire risks have developed significantly over the last ten years: The deterministic analyses are now solely based on CFD and the probabilistic analyses using Monte Carlo simulation have been carried out. These developments have improved the reliability and accuracy of the safety analyses. In addition, the same deterministic tools are also used for the analyses of external fire incidents, such as aircraft -impact induced fires. Maintaining the computational infrastructure (tools, hardware and competence) requires continuous investments, and significant research efforts are still needed to enable predictive engineering simulations of fire spreading.

FIRED-project will cover three main themes: fire risks of cables during the plant life cycle, fire Defence-In-Depth, and modelling tool development and validation. In addition, active participation to OECD PRISME 2 – project will continue.

1.1.1 Evaluating the risks during cable life cycle

Methods for predicting flame spread in cable trays have been developed in previous SAFIR programmes. They are based on the CFD simulation of the flame heat transfer and pyrolysis modelling, where the thermochemical degradation reactions are modelled using Arrhenius equation. Each reaction requires a set of parameters which need to be estimated from experimental small scale data [1]. The computational methods are developed in parallel to the experimental campaigns in USA [2] and within the OECD PRISME 2 –project in France.

Flame retardants such as aluminum trihydroxide and magnesium hydroxide are widely used in cable material formulations. Modelling these formulations became acute during the OL3 design and fire safety assessment, when the modelling methods had to be developed in a hurry, as these materials were already in the market and proposed by the reactor designers. New ingredients e.g. nanofillers / nanocomposites (organoclayes, mesoporous silicate particles, layer-by-layer technologies) are now emerging to the market, but the understanding of their flame retardant mechanisms on the level of modelling capability, and thus the competence for safety assessment is practically non-existent. In this project, we will pro-actively build a future competence for analysing the new flame retardants by carrying out fundamental studies of their performance and modelling.
The flammability of the electrical cable materials (mixtures of synthetic polymers and additives) is controlled by the reaction-to-fire tests according the international standards. After the installation, however, the polymer compositions will change, especially in the presence of oxygen, elevated temperatures and radiation. As these changes affect the mechanical and thermal properties, the ignition and flame spread characteristics after years of use can be quite different compared to the new cables. The current understanding is that the traditional PVC cables can actually become less flammable during the years of use, as the flammable softeners and other additives slowly escape from the polymer matrix. However, detailed ageing studies combining small scale experiments (micro-scale combustion calorimeter, thermogravimetry and cone calorimeter) and the flame spread tests with pyrolysis modelling have not been found. The studies within the SAFIR-programes have been limited to small and bench scale experiments of PVC cables with relatively small age difference and possible differences in initial compositions.

An interesting question is how the ageing affects the efficiency of the modern flame retardants that are now being installed (metal hydroxides) or those that are just emerging on the market (nano-fillers). Systematic experimental and modelling study with cable materials exposed to accelerated ageing is now proposed, improving the understanding of the impact of ageing on cable fire safety.

1.1.2 Atomistic modelling of novel flame retardants

The key physical and chemical processes of flame retardancy are connected to the thermal decomposition of an organic polymer at elevated temperatures. Flame retardant compounds modify the decomposition process to minimize the production of volatile fractions that would fuel flames, e.g. by means of thermal quenching. While a wealth of empirical knowledge exists on the mechanisms of flame retardancy, direct observations through atom scale simulations are, to the best of our knowledge, missing. Capability to reproduce the processes of flame retardancy in atomistic simulations would serve the purpose of continuum-level pyrolysis modelling in two ways. Firstly, reaction paths observed in the simulations could be used as a starting point for experimentally motivated pyrolysis models, such as those employed in plant scale fire simulations. This would be especially valuable when experimental knowledge on the reaction path is missing. Secondly, atomistic simulations could be used to predict both the reaction path and the associated chemical kinetics. This would enable constructing pyrolysis models entirely without experimental input—a useful capability for studying e.g. the effect of novel flame retardants on a variety of base materials. The idea of atomistically motivated pyrolysis modelling is illustrated in Figure 1.

Atom scale modelling of the mechanisms of flame retardancy is a demanding topic, as it requires the description of chemical reactions, i.e. the forming and breaking of chemical bonds, together with system size and time scales large enough for meaningful statistics and observation of the dynamic and reactive phenomena of interest. In practice, this means systems consisting of thousands of atoms simulated up to the nanosecond range. Ab initio quantum chemistry methods are the obvious choice for rigorous predictive chemistry, but their computational cost is prohibitive for the current needs. Classical Molecular Dynamics (MD), on the other hand, can handle systems with practical size and time scales, but a description of chemical reactions is missing. Reactive Molecular Dynamics (RMD) offers a trade-off between the efficiency of classical MD and the accuracy of ab initio quantum chemistry methods. Since the introduction of bond-order based reactive force fields for the classical MD framework in the late 1980’s ([3], [4]), the field of application for RMD has been growing steadily. In particular, the ReaxFF reactive force field [5] has proven highly popular. The set of chemical elements supported by ReaxFF is constantly expanding (e.g. [6]). This has, for the first time, enabled the RMD simulation of mineral flame retardants such as aluminium and magnesium hydroxide, as well as polymer-clay nanocomposites incorporating e.g. montmorillonite clay.

![Figure 1. Atomistic motivation for continuum-level pyrolysis modelling.](image)
1.1.3 Evaluating defence-in-depth

Defence-in-depth (DID) is one of the main concepts used to reduce the risk of severe accidents in nuclear facilities [7]. While the DID against large emissions of radioactive substances in case of core damage relies mostly on the use of physically nested systems and barriers, the DID against fires relies also on barriers and systems that are sequential in time. The levels of Fire-DID are traditionally considered to include [8]:

- Prevention of ignitions
- Fire detection
- Fire suppression by active systems and manually activated systems
- Physical confinement of fire effects by barriers
- Protection of sensitive equipment against fires

Most developments in the field have focused on assessing the performance of fire safety systems and equipment failures in the scope of the Fire-PRA. Technical methods to evaluate the individual levels of Fire-DID do exist and have been successfully developed and applied within the previous SAFIR-programmes and engineering analyses, but there is no systematic methodology to evaluate the realization of Fire-DID as a whole. There is a clear need for a means to evaluate the fire-protection within the entire operational environment, including e.g. regulatory framework, design methods, construction, safety culture, management, fire safety systems, structures and response procedures.

Concerning the individual technical aspects of the Fire-DID evaluation, the physical confinement by fire barriers was taken under investigation in SAFIR2014. Starting from the ideas of French EPRESSI-method [9], we developed a computational methodology to calculate the barrier failure probability [10] and an interface tool FDS2FEM [11] to enable analyses of fluid-structure interactions with CFD and FEM tools. To our surprise, we found that although the engineering methods for the load-bearing capability evaluation are well established and widely used, similar methods are scarce for the evaluation of insulation and integrity. Structural fire protection, which is the most traditional form of fire protection engineering, has the strong tradition of relying of fire resistance classifications, and this has delayed the development of engineering methods for barrier performance assessment, including the computational tools and, most importantly, the acceptance criteria. More work is needed to enable efficient assessment of structural and barrier performance within the fire simulation environment that is already commonly used for other purposes.

A specific need for the capability to carry out performance-based design of fire barriers comes from the renewal of the Finnish National building code, and the part E1 (Fire Safety of Buildings) in particular. In the renewal, which is expected to be ready by 2017, the amount of prescriptive requirements will be reduced and the potential for engineering application will most likely increase from the the current situation. This will affect the future NPP design and building processes.

The efficiency of Fire-DID evaluation process requires prior knowledge on the sensitivities of the system to the uncertainties associated with the physical boundary conditions, parameters and models. A specific type of a fire called ‘traveling fire’, where the region of combustion reactions moves over time, can cause high uncertainty to the local fire exposure. This phenomenon, first introduced after the WTC tower fire investigations, was also observed in the fire simulations of NPP cable room [12]. The possibility of traveling fires in other important spaces of the NPPs, such as reactor annulus, should be investigated.

In his study of German NPPs, Forell [13] found that the most significant factor affecting the reliability of fire barriers is the actual availability or presence of the barrier elements. The open doors or the lack of penetration seals were found to dominate the risk of fire spreading across compartment boundaries. The discussion about the relevance of this finding in Finland is still going on, and an immediate need for research cannot be confirmed.

One of the findings of the OECD WGRISK Fire-PRA workshop (April 2014) was that the coupling between Fire-PRA and the main PRA of the nuclear power plants is weak. The connection should be made between the fire scenario selection and the PRA process.

1.1.4 Modelling tools

Fire safety of existing or new installations is mainly investigated using numerical modelling. The modelling tools and their dependencies (data flows) are illustrated in Figure 2, where each text box represents one software type, with a name of the particular software given below the topic. The white boxes are computer programs that have been both developed and used in SAFIR programmes and the gray boxes are either commercial or open-source software developed by others.

In the centre of the modelling environment is the fire-CFD code Fire Dynamics Simulator (FDS), which is a low-Mach number solver for fire-driven flows [14]. It has a well-described Verification and Validation (V&V)
process which serves as a basis for software development Quality Assurance (QA) and provides an efficient framework for multi-site open-source development. An advanced feature of the FDS V&V process is the automated calculation of the uncertainty metrics that enable the quantification of the modelling uncertainty during the fire risk assessments.

FDS includes a specific module for human evacuation (FDS+Evac), which is also developed by VTT, but rarely used in the context of nuclear safety. Other CFD codes are occasionally used but the research focus in SAFIR programs is very much in the FDS-related issues.

FDS has the necessary sub-models for fire-related phenomena, many of which have been developed by VTT:

- Thermal radiation model of FDS is based on the Finite Volume Method for radiation with gray gas and scattering particles. The gray assumption of spectral dependence is sufficient in most scenarios, but may lead to inaccurate heat fluxes in case of high CO₂ and/or water concentrations. In addition, at long path lengths, even relatively modest concentrations of water vapour can cause an error in far-field radiation levels. In long term, the application of multi-band spectral methods should be considered.
- Liquid pool evaporation model: Pool fires are an important fire hazard in industrial environments. The research in Finland, up to this point, has focused on developing and validating the liquid pool evaporation models for fires in open atmosphere. There is a need for extension and validation of the current model for use in enclosures because liquid-fuelled fires at NPP’s often take place in compartments.
- The traditional way of modelling cable fire in FDS is via rectangular slabs, i.e. a simple class of immersed boundaries. These are relatively simple and fast to compute, but unfortunately lack accuracy in terms of flow between cables and geometry. For more accurate approach, sub-grid-scale particles has been considered. Cables can be modelled in cone calorimeter level as cylindrical particles, but the larger scale applications require more work. The targets of development are the interaction between particles, and their flow resistancy.
- The liquid spray model in FDS has been used to model both water-based suppression systems and the fast sprays resulting from impacts of liquid-containing missiles. Further validation of the model is required in order to quantify the uncertainties associated with the simulations of fireballs resulting from aircraft impacts. As a new topic, the penetration of aviation fuel inside the plant structures should be investigated.

Figure 2. Modelling tools for fire research and engineering.

Three different classes of program interfaces between the FDS and the other modelling tools can be identified:
1. Model input creation: The main research topic over the last few years has been the estimation of pyrolysis model parameters from small scale experiments. A specific tool PyroPlot has been developed [1]. The main estimation method is Genetic Algorithm (GA), which is an effective tool for estimating simultaneously several parameters from experimental data that may be noisy or otherwise complicated. The biggest drawback in GA is that the estimation is very slow. Attempts have been made to parallelize GA in PyroPlot, but the current version is far from optimal. PyroPlot has been used in past projects extensively, but its development and maintenance has been limited to make it work for each task at the time. PyroPlot has potential to attract users outside the project group if the necessary steps to improve the usability and documentation are taken.

The utilization of Reactive Molecular Dynamics for pyrolysis model input estimation has been envisioned, but the practical feasibility needs to be shown.

Two different alternatives are available for the classical graphical user-interface purpose. Neither of them have been used in the past projects, but they would offer a cost-efficient method to create complicated models, especially if the geometry can be transferred from existing design models in a digital form.

2. The deterministic simulations are used in Fire-PRA to investigate the possible fire outcomes at many different realizations of the input parameters. The parameter uncertainty is taken into account with Monte Carlo simulations. For this purpose, we have developed a specific tool Probabilistic Fire Simulator (PFS) [15]. A new development in this field is the modelling of organizational response, using the Stochastic Operation Time Model [16], which is formally a Monte Carlo simulation of an organizational graph. The recent experiences on using the model indicate that more flexible methods should be developed to predict the response instead of specifying it.

3. The third class of interfaces is the FDS2FEM that transfers fire model outputs into the boundary conditions of Finite Element Model of thermo-mechanical response of structures in the context of the DID. Other possible tools or methods are available for the same task but most of them have, up to date, been limited to very specific scenarios. The development of the generalized immersed boundary method within FDS will enable much more versatile geometrical treatments in the future. The development and maintenance of FDS2FEM should therefore be continued to support these new features.

Most of the fire simulations are made in PC or in a medium-scale cluster. Currently VTT has a Linux cluster (Smokey) that is exclusively reserved for fire simulations. It has 25 servers with a total of 256 cores and 720 GB memory. It is suitable for performing large, long or multiple simulations, but it is getting old, slow and too small for the current computation needs. Therefore investing in new cores is necessary.

Aalto University has access to two super-computers (Sisu and Taito) at CSC IT Centre for Science, although any massively parallel simulations are currently not envisioned, preventing the utilization the latest Sisu supercomputer. Both serial and parallel FDS simulations, including the complete FDS verification suite, have successfully been run at Taito.

1.2 Objectives and expected results

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. To meet this main objective, the following technical or detailed objectives are specified:

1. Development of a pyrolysis modelling capability for the new flame retardants for predicting their performance in nuclear power plant fire scenarios. The expected outcomes are
   a. Evaluation of the usability of Reactive Molecular Dynamics (RMD) in modelling the flame retardant mechanisms.
   b. Application of RMD to provide reaction schemes and kinetic parameters for one or two flame retardant systems.
   c. Pyrolysis model reaction schemes and parameters for application in fire CFD.

2. Quantification of the ageing effect on modern cables through multi-scale experiments and modelling. The expected outcomes are
   a. Effect of accelerated ageing on the chosen flame retardant.

3. Development of a capability to predict the fire resistance of a barrier element with the Fire-CFD and – when necessary – the 3D-FEM tools. The fire resistance should include the aspect of load bearing (R), integrity (E) and insulation (I). The expected outcomes of the project are
   a. FDS submodel for barrier fire resistance calculation.
b. FDS2FEM support for immersed boundary method.
c. Risk-based acceptance criteria to replace/supplement those used in the standard testing.

4. Exploring the **wider context and possible implementations of the Fire-defence-in-depth** concept.

The expected outcomes are

a. Understanding of the “interdisciplinary depth” at which the fire safety objectives should be taken into account during a plant design. For example, if the likelihood of an unwanted outcome is calculated using an event or fault tree, the DID levels are seen as sequential levels of the tree. The *interdisciplinary depth* is related to the value of each nodal probability, i.e. the reliability of each DID level. It could be measured as an explicit level of engineering competences, design requirements, organizational or management system specification or ethical norms that affect the node probability.

b. Analysis of a simple test scenario for the possible factors of Fire-DID.

c. Better integration of fire analyses and Fire-PRA into main NPP PRA and traceability of fire analysis inputs.

5. Continuous **development and maintainence of the fire modelling tools** to meet the needs arising from the increasing community of end-users, to maintain the simulation competence, and to solve the found issues and problems of the software. The expected development outcomes are

a. FDS: Validated liquid pool evaporation model for fires in closed or mechanically ventilated compartments. (Continuation from SAFIR2014).

b. FDS: Improved reliability and user guidance for the Lagrangian particle –based description of the electrical cable fire source. (Continuation from SAFIR2014).

c. FDS: New particle interaction model enabling surface cooling of fuel particles (cables) by liquid particles (water droplets).

d. FDS: Validation study of fuel sprays to increase the confidence of the aircraft impact fire simulations.

e. PFS: Support for large-scale parallel computing environment and the necessary guidance.


6. Participation in **steering and utilization of the on-going OECD PRISME2** –project. The expected outcomes are

a. Project progress reports in the Reference Group meetings.

b. Transfer of project results to the researchers and other Finnish organizations.

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 3.
1.3 Exploitation of the results

The results of FIRED will be utilised by a versatile group of experts during the future applications and research projects. The means of exploitation of the FIRED outcomes are summarized in Table 1. The capability to evaluate flame retardants will be mainly exploited by researchers, but on the demand from the authorities. The DID evaluation capability, in turn, will mainly be exploited by the authorities, although the actual studies of fire scenarios will be used by the utilities in fire-PRA. The researchers will benefit from gaining knowledge about novel FR mechanisms and improving the computational capability for future needs. The new and improved modelling tools will be utilised by fire safety engineers and the whole fire safety community around the world.

Table 1. Means of exploitation of the results in FIRED. Short = 1-2 years, Mid = 2-5 years, Long ≥ 5.

<table>
<thead>
<tr>
<th>Task</th>
<th>Result</th>
<th>Who</th>
<th>How</th>
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<td>1.1</td>
<td>Feasibility of RMD for FR studies</td>
<td>Research</td>
<td>Kick-off for a completely new research track.</td>
<td>Short</td>
</tr>
<tr>
<td>1.1</td>
<td>Increased understanding of new FR mechanisms</td>
<td>Research</td>
<td>Risk assessments and material R&amp;D</td>
<td>Long</td>
</tr>
<tr>
<td>1.1</td>
<td>Pyrolysis modelling of new FR cables</td>
<td>Research</td>
<td>Risk analyses</td>
<td>Mid</td>
</tr>
<tr>
<td>1.1</td>
<td>Application to plant scale risk assessment</td>
<td>Authorities Utilities</td>
<td>Input for PRA</td>
<td>Short</td>
</tr>
<tr>
<td>1.2</td>
<td>Aging effect of new FR cable materials</td>
<td>Utilities Authorities</td>
<td>Life-cycle management</td>
<td>Mid</td>
</tr>
<tr>
<td>2.1</td>
<td>CFD-simulation tool for fire barrier response</td>
<td>Research Utilities FSE</td>
<td>Performance-based design of compartmentation, risk analyses</td>
<td>Mid</td>
</tr>
<tr>
<td>2.2</td>
<td>Fire-DID interdisciplinary aspects</td>
<td>Research Utilities</td>
<td>Widening the scope and practical utilization of DID</td>
<td>Mid</td>
</tr>
<tr>
<td>2.2</td>
<td>Reliability estimate of DID fulfilment</td>
<td>Authorities Utilities</td>
<td>Risk analyses</td>
<td>Long</td>
</tr>
<tr>
<td>2.3</td>
<td>DID consideration in Fire-PRA</td>
<td>Utilities</td>
<td>PRA improvement</td>
<td>Long</td>
</tr>
</tbody>
</table>
### 1.4 Appropriateness of the project to SAFIR2018 programme

The proposed project has contact points with all three research areas described in the SAFIR2018 framework plan and supports strongly the goals of the SAFIR2018 programme. In addition to these substantial contact points, the project supports the goal of the framework plan by educating new experts in the area of fire safety in nuclear power plants.

The first WP, cable fire risks during plant life cycle, has two sub-tasks: impact of cable ageing and new flame retardant polymers. The subject of cable ageing relates to the research need estimating failure mechanisms of components in nuclear power plants in research area Structural safety and materials. The second one, new flame retardant polymers as cable materials, relates to the topic “New material solutions” in the same research area.

The second WP of the project, fire-barrier performance assessment, deals with the concept of defence-in-depth which is a central part of the research need concerning overall safety in research area Plant safety and systems engineering. The fire barrier performance is also closely related to the research need on advanced methods for estimating structural safety in research area Structural safety and materials. As this research area also deal with Probabilistic Risk Analysis (PRA), the sub-task Fire-PRA integration in plant PRA is in connection with the Reactor research safety area, but also with the research need on interface between PRA-based and deterministic planning in research area Structural safety and materials.

In the third WP, fire simulation development, maintenance and validation, the softwares Fire Dynamics Simulator (FDS) and Probabilistic Fire Simulator (PFS), among others, are further improved. These subjects are in clear connection to the research area Reactor safety where one of the main tasks is the development of computational facilities to ensure that the plant and its systems fulfil the safety requirements.

The fourth WP, PRISME2 participation and utilization, concerns the international OECD PRISME 2 project, studying the spreading of heat and smoke in enclosure fires, spreading of fires on cables and fire suppression by water based systems. As noted in the framework plan, international co-operation is a necessity in nuclear safety research, especially for small actors like Finland. This task is a continuation of previous participation in PRISME2 and its predecessor PRISME, which have successfully transmitted knowledge and results to interested parties in Finland.

### 1.5 Education of experts

The project task 1.1 “Atomistic modelling of novel flame retardants” contributes to Mr. Antti Paajanen’s doctoral dissertation, supervised by prof. Kai Nordlund (University of Helsinki). Working title of the thesis is “Reactive molecular dynamics studies on the high-temperature behaviour of soft condensed matter”.

The FDS development and validation studies (task 3.1) will contribute to the doctoral dissertation of Mr. Topi Sikanen at VTT during the project period 2015-2018, supervised by prof. Simo Hostikka (Aalto University). The work has started in the previous SAFIR-programme.
Part of the research will be carried out at the Aalto University’s department of Civil and Structural Engineering. The project will strengthen the Finnish expertise in structural fire safety and fire simulation by educating the civil engineering students. Aalto University will recruit two young researchers or research trainees to work in the field. One of them will focus on the defense-in-depth aspects and the structural performance – a topic that is in the core of the department’s curriculum. The results will contribute to studies leading to Master and Doctoral level. The person will also work as a teaching assistant at the department, thus sharing the gained knowledge. Another student will participate in the development and maintenance of the probabilistic simulation tool, giving a student an excellent view to the scientific computing in the context of fire risk analysis.
2. Work plan

FIRED-project has four main topics, which form the four work packages of the project:

WP1. Cable fire risks during plant life cycle,
WP2. Fire-barrier performance assessment,
WP3. Fire simulation development, maintenance and validation,

Planned person months and task structure is presented in Table 2. The topics and work plans for 2015-2018 will be presented in more detail in following chapter. The work will be done in cooperation between VTT and AALTO.

Table 2. Work plan and person months of FIRED for 2015-2018. Blue – VTT, Red – AALTO, Green – Both. The changes in the budget of 2017 are highlighted with red background.

<table>
<thead>
<tr>
<th></th>
<th>2015</th>
<th>2016</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
<tbody>
<tr>
<td>WP1 - Fire risks during cable life cycle</td>
<td>5</td>
<td>4</td>
<td>3.5</td>
<td>4</td>
</tr>
<tr>
<td>T1.1 New flame retardant polymers</td>
<td>3</td>
<td>2</td>
<td>3.5</td>
<td>4</td>
</tr>
<tr>
<td>T1.2 Impact of cable ageing</td>
<td>2</td>
<td>2</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>WP2 - Fire-barrier performance assessment</td>
<td>7</td>
<td>9</td>
<td>7</td>
<td>7</td>
</tr>
<tr>
<td>T2.1 Barrier performance assessment with Fire-CFD</td>
<td>7</td>
<td>9</td>
<td>6</td>
<td>6</td>
</tr>
<tr>
<td>T2.2 Interdiciplinary depth of defence</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>T2.3 Fire-PRA integration</td>
<td>0</td>
<td>0</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>WP3 - Fire simulation development, maintenace and validation</td>
<td>6</td>
<td>3</td>
<td>2.5</td>
<td>11</td>
</tr>
<tr>
<td>T3.1 FDS development, maintenance and validation</td>
<td>4</td>
<td>3</td>
<td>2.5</td>
<td>8</td>
</tr>
<tr>
<td>T3.2 PFS development and maintenance</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>3</td>
</tr>
<tr>
<td>T3.3 PyroPlot development and maintenance</td>
<td>2</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>WP4 - PRISME 2 and 3</td>
<td>1</td>
<td>1</td>
<td>3</td>
<td>1</td>
</tr>
<tr>
<td>T4.1 Participation fee of PRISME2 or 3</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T4.2 Utilisation of results</td>
<td>1</td>
<td>1</td>
<td>3</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>19</td>
<td>17</td>
<td>16</td>
<td>23</td>
</tr>
</tbody>
</table>

Participation to international meetings and conferences is an important part of the dissemination of the results and international cooperation. Meeting plan is shown in Table 3, and it is assumed that the travel costs are maximum 2 000 € / travel / person.

Table 3. Travelling plan for FIRED 2017.

<table>
<thead>
<tr>
<th>Meeting</th>
<th>Organisation</th>
<th>Number of participants</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRISME2 final seminar</td>
<td>VTT, AALTO</td>
<td>2</td>
</tr>
<tr>
<td>PRISME3 meeting</td>
<td>VTT</td>
<td>1</td>
</tr>
<tr>
<td>2 times a year</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Conference</td>
<td>AALTO</td>
<td>1</td>
</tr>
</tbody>
</table>
2.1 Work package 1 (WP1)

This work package studies the cable induced fire risks during plant life cycle, including the performance of new flame retardant cables and their ageing. The emphasis is in the numerical simulations and model development, but significant amount of experimental work will also be performed to support the modelling. WP1 is divided into two tasks. The first one deals with novel flame retardants and their multi-scale modelling, and the second covers the performance of aged cables.

Task 1.1 will cover the modelling of novel flame retardant cable materials. A new approach of using atomistic simulations is used here in two phases. The first one is a proof-of-concept phase. RMD simulations will be used to investigate the effect of a chosen, well-known, flame retardant on the decomposition pathway of a representative polymer used in cable jackets. The goal is to provide a detailed atom-level description of the relevant flame retardant mechanism, which has to be consistent with the mechanism proposed in the literature. Small-scale experiments, such as TGA combined with analytics for volatile compounds, can be used to support the model drafting process. The task will proceed into the second phase only if the proof-of-concept gives positive results.

In the second phase, the RMD simulation approach will be used to construct a continuum-level pyrolysis model for a polymeric material relevant for cable applications. Both the decomposition pathway and the associated kinetics will be drawn from the atomistic simulations. The feasibility of the resulting pyrolysis model will be assessed using FDS—by comparing TGA and cone calorimetry experiments against corresponding simulations.

More traditional pyrolysis modelling will be performed according to the results of the atomistic studies. These models will also be used in a case study fire simulation during the last year of FIRED (2018). Supporting small and bench scale experiments will be performed for selected cable samples.

Task 1.2, the ageing studies, aims to an understanding of the impact of ageing on the fire performance of electrical cables of different materials relevant for nuclear power plants. The outcome of the task is improved ability to estimate fire risks due to ageing changes in cable material properties during the whole life-cycle of the plant. The task starts with a survey on new cable materials and choice of cable samples, planning of well-designed accelerated aging and experiments on non-aged and aged samples. Polyvinyl chloride based cables, which are widely used in present plants, can be used as a “baseline” for fire performance of cables. More recent cables using metal hydroxides as fire retardants are one possibility, and cables containing new ingredients like nanofillers may represent modern cable technology.

The accelerated ageing is carried out at elevated temperature for a certain time, where both temperature and time are to be estimated to correspond to a time period of normal plant circumstances. The experimental series include small scale experiments with micro-scale combustion calorimeter, thermogravimetry and cone calorimeter and medium-scale flame spread tests with the 2-m apparatus developed at VTT. Properties studied in the experiments include time to ignition, rate of heat release, heat of combustion and rate of flame spread. Results compare the aged cables with unaged and also different cable materials with each other. Attempts are also made to apply modelling methods to the ageing phenomena of electrical cables analogous to the pyrolysis modelling carried out at VTT.

Partners and person months allocated to WP1 to be given in the table.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months 2015-2018</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>16.5</td>
<td>3.5</td>
</tr>
<tr>
<td>AALTO</td>
<td>0</td>
<td>0</td>
</tr>
</tbody>
</table>

2.1.1 New flame retardant polymers (T1.1)

The proof-of-concept results of MD studies in the previous years have exceeded all the expectations. The MD concept is able to capture the flame retardant mechanism in a polymer, and additionally to provide novel insight to the phenomenon which can already be used for continuum level modelling.
The research activities related to the atomistic modelling of novel flame retardants – task will resume in 2017. The analysis of the proof-of-concept phase simulations (performed during 2016) revealed that the existing database is too narrow to draw definite conclusions on the nature of the decomposition kinetics. Analysis suggested that the decomposition of pure ATH is diffusion-controlled, which is surprising given that diffusion usually is the governing mechanism when two physically separate reactant masses interact. The temperature dependence of the data could not be explained by single activation energy for decomposition. Further decomposition runs for pure ATH are required for different system sizes and a number of temperatures to resolve the open questions.

The analysis of the existing decomposition data for ATH-HDPE system confirmed the presence of a chemical component of ATH on the HDPE pyrolysis: the water yield at high temperatures could only be explained such that hydrogen from HDPE is present in the product molecules. At the same time, the chemical component brought an unexpected complication to the analysis of the decomposition kinetics, since instead of a pure decomposition reaction to produce water, a second pathway exists through chemical reactions between oxygen from ATH and hydrogen from HDPE. Resolving this complication will require a large number of decomposition simulations for the ATH-HDPE system at various configurations and temperatures. In addition, the decomposition of pure HDPE needs to be studied as the base case. A special feature of real HDPE is that it is not a well-defined material, but it has areas of crystalline and amorphous regions. Therefore, in all simulations involving HDPE, the degree of crystallinity needs to be taken into account.

Towards the end of the project, the RMD simulation approach will be used to construct a continuum-level pyrolysis model for the ATH-HDPE system. Both the primary decomposition pathways and the associated kinetics will be drawn from the atomistic simulations. Feasibility of the resulting pyrolysis model will be validated against small-scale experiments.

2.1.2 Impact of cable ageing (T1.2)

This topic won’t be active in 2017 as the management board of SAFIR2018 has decided that testing of irradiated materials will not be topical in 2017.

2.2 Work package 2 (WP2)

The second work package develops the assessment methods for fire-defense-in-depth and explores means to improve the fire safety through interdisciplinary consideration of fire safety in different processes affecting the failure probability of the sequentical fire-DID levels (ignitions, detection, suppression, barriers, compartments). In 2015 and 2016, the work has focussed on the development of new, finite-elements based heat conduction solver for the condensed-phase analyses of FDS, and the evaluation of model uncertainty in the chain of modelling (2016).

In task 2.1, the computational tools will be developed for the prediction of the functional barrier fire resistance within fire-CFD (FDS). The previously developed interoperability tool FDS2FEM will be developed further to improve the support for new computational techniques of complicated geometries (immersed boundary method). Acceptance criteria of barrier performance will be formulated by reviewing the damage criteria of protected devices and components.

In addition to the technical tools, in task 2.2 we will study the possibilities to extend the DID approach from physical systems to a wider context, including the design requirements and operational instructions. We will attempt to define interdisciplinary measures for the degree or quality of fire safety considerations supporting the reliability of the physical systems. The work will first review the safety and security principles from other fields of industry, collect the aspects of nuclear power plant safety that can and should be taken into account, and finally try to quantify the impact of various factors on the DID level failure probability. For instance, how do the current regulations and norms, working habits and plant operations affect the reliability of fire detection system?

Finally in task 2.3, we will investigate the relationship between Fire-DID, Fire-PRA and the main PRA. The technical and non-technical developments are combined with the fire-PRA technique in a manner similar to human reliability assessment (HRA). This time, instead of introducing a number for the human error probability, we will introduce a number for the DID level failure probability, possibly replacing conservative assumptions. The work of this task will be carried out jointly with another SAFIR2018 project (PRAMEA proposal Task 4.1).

Partners and person months allocated to WP2 to be given in the table.
2.2.1 Barrier performance assessment with Fire-CFD (T2.1)

This task will be implemented as a collaborative project with Lund University, Danish Institute for Fire and Security (DBI) and Vattenfall Rinhals (Sweden), under the NKS-project FIREBAN. The following FIREBAN workpackages include significant contributions from Aalto University and VTT:

FIREBAN WP 1: State-of-the-art for fire barrier reliability assessment.
We will collect the state of the art on methods and experiences to determine the reliability of fire barriers. We will review of validation data within open literature and other projects.

We will determine the relationship between the standard-fire based fire resistance classification and the failure risk under real fire conditions and real protection objectives. First, the risk-based acceptance criteria will be established for insulation, integrity and stability, considering the different spaces to be protected. Next, we will determine a number of risk-based fire curves for NPP rooms based on real fire scenarios.

FIREBAN WP 3: Reliability determination
This work will contain four major routes of determination:
A. Use of existing test fire test data in combination with new test data
B. Use of statistics
C. Use of modelling for the fire barrier performance assessment.
D. Practical test design

FIREBAN WP 4: Dissemination of results
The practical research topics of 2017 include
- the combination of deterministic modelling uncertainty of wall temperatures with the propagation analysis, developed in 2017, thus demonstrating and testing the practical application of the propagation analysis,
- testing the 3D heat conduction solver developed in FDS against separate CFD+FEM approach.

2.2.2 Interdisciplinary depth of defence (T2.2)

This topic is not active in 2017.

2.2.3 Fire PRA integration

VTT will organise a workshop on the Fire PRA, and its relation to the Fire-DID and the main PRA. This one day workshop will be organized in Espoo. The workshop will focus on:
- Development of deterministic analyses, including their verification and validation, and their applicability on the Fire PRA,
- Present status of Fire PRA on Finnish plants and the near future plans to upgrade them,
- Main PRA and its future plans: Development work needed to address the main PRA and Fire PRA interface.

Participants of the workshop are mainly from the NPP companies, Finnish authorities, and research organizations. One objective of the workshop is to create a roadmap for the Finnish Fire PRA development work.

2.3 Work package 3 (WP3)

VTT has developed several models and tools for simulation and analysis of fire risks. The third work package is devoted to continuous development and validation of these tools. The major tasks in this work package are related to validation and utilization of models developed previously.

Task 3.1 will cover the development and validation of FDS. We will add a new model into FDS to calculate surface cooling of particle-based cable models by liquid sprays. This requires development of novel computational techniques of particle-particle interactions. The previously developed pool fire model will be developed further and
validated for use in compartment fire scenarios (PRISME and PRISME 2 data). Further validation work on the spray model in FDS is will be carried out in order to increase the confidence of aircraft impact simulations. The development of Lagrangian particle-based cable models will also continue.

Task 3.2 is about improvement and maintenance of Probabilistic Fire Simulator (PFS). The current version will be updated to operate with the latest FDS version (PFS contains a simple FDS-interface). The previously developed operation time model will be integrated to PFS, and the implementation of agent-based operation simulations through PFS will be investigated. Agent-based simulations represent a completely new approach towards organizational simulations. This task is very much educational as new experts to perform Monte Carlo –simulations are needed.

Task 3.3 will develop and maintain PyroPlot. The overall goal is to transform the current PyroPlot that is mainly used as a GA tool by few people to a more user-friendly, general and open tool that fire researchers and engineers can use in their work. The first step is to improve the parallelisation of the genetic algorithm to allow more efficient parameter estimation. The second priority is to make the user interface more intuitive and user friendly, and update the user’s guide to reflect the current capabilities of the program. In addition, the tool should be generalised for reading any experimental results, and more options for algorithms and methods should be included. The tool is implemented with Matlab.

Partners and person months allocated to WP3 to be given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months 2015-2018</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>19.5</td>
<td>2.5</td>
</tr>
<tr>
<td>AALTO</td>
<td>3</td>
<td>0</td>
</tr>
</tbody>
</table>

2.3.1 FDS development, maintenance and validation (T3.1)

In 2017 the FDS development will mainly focus on the cable modelling. The objective of the predicting flame spread in a cable tray will be approached from two directions: Solving the current issues in the particle models of cables, and improving the methods of pyrolysis modelling that is the basis of the predictive flame spread simulations. The work started in 2015, regarding the modeling of cable trays using lagrangian particles will continue. The work will focus in the particle-particle interactions and modelling correctly the flow drag caused by cables and the heat transfer to cables in a sub-grid scale cable tray model.

2.3.2 PFS development and maintenance (T3.2)

This task is not going to be active in 2017 due to the limited financial resources.

2.3.3 Pyroplot development and maintenance (T3.3)

This task is not active in 2017.

2.4 PRISME 2 (WP4)

OECD PRISME 2 project was started in 2011 and, according to the current schedule (Figure 4), it will end in 2016. The last reports are due in 2017. The project has bi-annual meetings of Program Review Group (PRG) and Management Board. Additional Analytical Working Group is formed to perform fire model validation on voluntary basis. It gathers usually one day before the PRG. VTT is the official signatory of the project from Finland.

By active participation, Finland has been able to convince the other partners and the operating agent (IRSN) to accept our proposals in many decisions concerning the project. For instance, the TVO’s cables were used as a fire source in the PRS2-CFS campaign, providing information about cable tray burning rates that is directly applicable to Finnish plants. For the fire extinction experiments (FES), we were able to specify the test program in a way that best serves the needs of the modelling without compromising the value for practitioners. The decisions of the last campaign will be made in November 2014. In this project, results from PRISME2 are utilised mainly in
WP3. The PRISME projects provide a comprehensive, well instrumented experiment series for use in validation and development of computational models.

The remaining time of the PRISME 2 will focus on the analysis, utilization and dissemination of the experimental results. Active participation will be important to ensure that the project deliverables provide the information that is important for the Finnish beneficiaries, and that they can be efficiently exploited by the Finnish experts both in power companies and research institutes.

![Figure 4. OECD PRISME 2 schedule.](image)

Partners and person months allocated to WP4 to be given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months 2015-2018</th>
<th>Person months 2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
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<td>3</td>
</tr>
<tr>
<td>AALTO</td>
<td>2</td>
<td>0</td>
</tr>
</tbody>
</table>

Next experimental programme called PRISME3 will start in the end of 2016. The details of the project plan are provided in the next sub-section.

2.4.1 PRISME3 participation (T4.1)

PRISME3 will start in the end of year 2016 and last until 2021. Finland’s share of the participation fee is 100 k€ during 5 years, 20 k€ / year. The experimental campaigns of PRISME3 are:

- CAMPAIGN 1: Smoke stratification and SPREAD (PRISME3-S3) 2017
- CAMPAIGN 2: ELECTRICAL cabinet fire SPREAD (PRISME3-ECFS) 2018-2019
- CAMPAIGN 3: Cable Fire Propagation (PRISME3-CFP) 2019-2020
- CAMPAIGN 4: Complementary tests (PRISME3-COMTE) 2020-2021.

Finland’s participation to PRISME3 was discussed in RG2 meeting in October 2016 (and in several earlier meetings). The topics of the new programme are considered very interesting and relevant for the NPP’s. Especially campaign 2 considering electrical cabinet fires, and campaign 3 about cable fire propagation raised interest among the reference group. The results of the electrical cabinet fires can be directly applied in the safety analysis at the NPPs. The results of cable fire propagation are important especially for model validation, but also in direct applications. The plan includes a test of 10 m long cable tray, simulating a cable tunnel. This kind of test has not been done before.
2.4.2 PRISME results utilisation (T4.2)

We have participated to two five-year programs of OECD/NEA PRISME projects during the years 2006-2016, and the third experimental program is starting in the end of 2016. During these years, we have gained loads of experimental data to be utilised. Some part of the data has already been used either directly in the safety assessments of a NPP or in model development and validation. The volume of the utilisation work has been minor compared to the possibilities.

In 2017 we will dedicate some persons months for utilising the PRISME results, starting from the cable fire spread. This work will contribute to training and academic education of a research trainee. The work will continue in 2018, and the results of PRISME3 will be taken into the analysis when they are ready (unless if Finland decides to not participate to PRISME3).

The scientific and technical novelty of the PRISME results up to this date are related to the characterization of pressure and ventilation flows in a mechanically ventilated and well-controlled environment. The research has resulted in a vast amount of experimental data which has been used to validate FDS capability to predict ventilation flows, both within the FDS Validation database [17] and as scientific publications [18].
## 3. Deliverables 2016

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Atomistic modelling of novel flame retardants (manuscript of a journal article)</td>
<td>3.5</td>
<td>15.12.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Computational methods to evaluate the performance of fire barriers with reactive and layered materials</td>
<td>6</td>
<td>15.12.2017</td>
</tr>
<tr>
<td>D2.3.1</td>
<td>Fire PRA workshop</td>
<td>1</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Validation and verification of sgs particle drag and heat transfer, manuscript of a journal article</td>
<td>2.5</td>
<td>30.09.2017</td>
</tr>
<tr>
<td>D4.2.1</td>
<td>Utilisation of PRISME results - report</td>
<td>3</td>
<td>15.12.2017</td>
</tr>
</tbody>
</table>

| Total pm          | 16                                 |
4. Project organisation

The project organisation consists of researchers in VTT and Aalto University. Project manager of the whole project is Dr. Anna Matala (senior scientist) from VTT, and Professor Simo Hostikka is responsible of the work of Aalto. Most of the tasks have been nominated to one single partner (VTT or Aalto), but task T4.2 will be carried out in cooperation. Naturally exchange of knowledge and ideas will take place between the two organisations in all tasks.

An NKS joint project FIREBAN (2016-2018) is integrated into the FIRED-project, as described in Section 2.2.1. The list of the main researchers, tasks and the estimated person months for 2017 are listed in Table 4. VTT will consider the possibility of hiring a young researcher or a PhD student for WP4 (utilisation of PRISME results).

Table 4. List of main researchers in 2017.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Anna Matala</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>T3.1</td>
<td>0.5</td>
</tr>
<tr>
<td>Topi Sikanen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T3.1, 4.2</td>
<td>2</td>
</tr>
<tr>
<td>Antti Paajanen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>0.5</td>
</tr>
<tr>
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5. Risk management

Most of the risks identified for FIRED project plan are related to the personnel resourcing (hiring or other work load of the key persons) and to the experimental equipment and scheduling. For the modelling work it is obligatory to have the necessary experimental results ready before the modelling can happen. The risk assessment of the project management is a continuous task in FIRED. All potential delays and risks are discussed in project group, and compensatory plans will be made for minimizing the consequences. The identified risks of FIRED are listed in Table 5.

Table 5. Critical risks for implementation.

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<th>Proposed risk-mitigation measures</th>
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<td>Delays in material delivery</td>
<td>WP1</td>
<td>Cables may be obtained from NPPs, purchased, or asked directly from the manufacturer. In case of delay in one supplier the others will be asked. The product amounts are not high, and should not cause any problems.</td>
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<td>(cables).</td>
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<td>Unavailability or loss of</td>
<td>WP1</td>
<td>Careful planning in the beginning of the year to make sure the experimental equipments are available when needed. In case of total loss of equipment, reallocation of resources.</td>
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<td>experimental equipment.</td>
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<td>Breaking of experimental</td>
<td>WP1</td>
<td>Caution is taken when performing experiments with sensitive equipment. Some relevant spare parts have been purchased beforehand for ensuring fast recovery.</td>
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<td>equipment.</td>
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<tr>
<td>Topics not suitable for students.</td>
<td>WP1, WP2</td>
<td>Research topics are selected so that many approaches can be used to investigate the problem.</td>
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<td>Delays in recruiting (AALTO).</td>
<td>WP2</td>
<td>The envisioned amount of work for the research trainee / MSc student during the first year does not require a full year attendance, thus giving few months time to find a suitable person in 2015. The continuity from MSc towards doctoral studies is emphasized early on.</td>
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<tr>
<td>Lacking computational resources.</td>
<td>WP3</td>
<td>Model development and especially validation simulation require significant amount of computational time. VTT’s Linux cluster will be updated for better performance. The CSC resources are also available for use (although no separate budget has been reserved at this point.)</td>
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<tr>
<td>Balance between practical</td>
<td>All WP</td>
<td>The topics and results will be discussed with representatives of authorities and end-users before, during and after the research, and their opinions are valued and taken into account.</td>
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<td>applications and basic research</td>
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<td>that will benefit in future.</td>
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<td>Changes in project group</td>
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<td>This is a challenge if each competence is mastered by one person. This is prevented by transferring knowledge in project group and sharing the work. Research trainees and students will be hired and educated as new experts.</td>
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<td>Reallocation of resources and/or prioritising most significant tasks.</td>
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References


## Work packages and Tasks

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SAFIR 2018 Project plan

FOUND
Analysis of Fatigue and Other cUmulative ageing to exteND lifetime

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3.2.2017
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1. Research theme and motivation

The 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.

The focus areas are as follows:
- 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 and transferability; beginning with a master’s thesis.
- WP4: Fatigue and crack growth caused by thermal loads, with emphasis on modelling, and including a licentiate thesis.
- WP5: Development of RI-ISI methodologies, including participation to ENIQ Task Group Risk (TGR) activities.
- WP6: Dynamic loading of NPP piping systems.
- WP7: Residual stresses in BWR NPPs.

1.1 Background and state-of-the-art

The background and state-of-the-art review of the topic of each work package are presented in the following subsections.

1.1.1 WP 1 - Remaining lifetime and long term operation of components having defects

With the modern developing NDE methods more indications are found year by year. The computational assessment of flaw/crack behaviour due to fatigue or stress corrosion cracking (SCC) under operational loading including residual stresses is important for determining remaining lifetime of components.

In the lifetime assessment of components numerical methods for computing welding residual stresses and subsequent operational stresses are in practical use. There is uncertainty, however, concerning how to computationally assess the possible flaw/crack behaviour under the computed residual stress field and under the subsequent operational stress state. The numerical computation for assessing the crack criticality and possible growth is not straightforward due to the limitations of common methods to evaluate parameters describing crack loading.

The present capabilities for the computation of the time and/or load cycles to reach the maximum allowable crack sizes, as determined by the ASME code [1.1], are based mainly on engineering solutions, leading often to conservative results. Other engineering methods such as the R6 method exist as well and can be utilized in the NPP component assessments with reasonable justifications. The relation between different methods is not fully established and a FAD based approach might be beneficial over ASME rules.

1.1.2 WP 2 - Susceptibility of BWR RPV internals to degradation mechanisms

This work package concerns research on the susceptibility of boiling water reactor (BWR) reactor pressure vessel (RPV) and its internals to various relevant degradation mechanisms. The results of the work will be reported in a dissertation. The ageing degradation of NPPs emphasises the need to anticipate the possible degradation mechanisms. This is an important issue both domestically and abroad. Then, the corresponding plant experiences from other countries are one important data source. On the other hand, published research results also offer valuable data on these issues. Recent relevant research publications on degradation of BWR RPV internals include those by EPRI [2.1], NRC [2.2] and IAEA [2.3]. The necessary new developments include BWR internals specific
review of susceptibility to degradation mechanisms. This is possible through co-operation with Finnish power company TVO, who will provide the necessary technical plant information.

The relevant degradation mechanisms affecting BWR RPV internals and their supporting structures include e.g. radiation embrittlement, thermal embrittlement, fatigue, stress corrosion cracking (SCC) and erosion-corrosion. Representative examples of BWR RPV internals include flange cooling spray piping, steam separator supports, feedwater spargers, control rod guide tubes, and moderator tank.

1.1.3 WP 3 - Fatigue usage of primary circuit

In 2002 the YVL 3.5 guide [3.1] introduced requirements on accounting for environmental effects in fatigue: “In the fatigue assessments the effects of environmental conditions shall be taken into account. Coolant oxygen content, operational temperature, impurities of the applied materials and the loading induced strain rate of the material all affect the fatigue life of the materials used in the primary circuit.” Similar requirements have been set e.g. 2006 in Japan for plant life management [3.2] and 2007 in USA for new designs and life extensions [3.3]. In response to these regulatory requirements and anticipated research needs in Finland, an advanced FaBello test equipment capable to perform direct strain controlled tests with standard specimens [3.4] was designed. A working prototype was achieved in the Safir2010 programme [3.5]. Since 2010, FaBello has been further developed and used to support an update of the German fatigue design rules [3.6-3.8].

The performance of inexpensive sensors used for strain measurement in FaBello has been unsatisfactory, but new “space technology” sensors have become available and are currently being installed to FaBello. Assuming that they finally support straightforward calibration, FaBello units can finally be considered to represent globally most advanced state of the art in experimental technology for LWR environmental fatigue, ready to generate experimental data in support of a mechanism informed corrosion fatigue model (as part of research in SAFIR programme) or fatigue assessments for NPP primary circuits (not within SAFIR). FOUND project will help in maintaining this capability and train a researcher to utilise it.

The importance of environmental effects in fatigue performance of primary circuit materials is amplified by lack of validated data, unknown mechanisms, general misunderstandings and disagreement. Interest on this topic continues and new test facilities are being developed in Europe. However, VTT is the only laboratory applying test methods, which are directly compatible with the code curves [3.9,3.10]. Areva NP GmbH (Erlangen) is aiming to develop a similar capability. The Argonne National Laboratory data [3.1] for U.S. NRC consist mainly of (over the autoclave measured) displacement controlled solid axial bar tests. MPA in Germany has used similar arrangement. Japanese laboratories introduced a standard specimen with the shape of a thin walled tubular specimen, where water is circulated inside and extensometer is applied outside the tube. Also hourglass type specimens, where axial strain is estimated based on diametric changes, are used [3.12]. EDF R&D (France), Amec (U.K.) and some other European laboratories target in tubular specimen facilities, which may be good for scientific studies, but are not “design compatible” to directly validate fatigue assessments. Indirect methods to estimate strain may work satisfactorily for ferritic steels, but for austenitic stainless steels, which are used in Finnish NPP primary circuits. Complex cyclic response of stainless material grades corrupt correlations between displacement and strain.

Lack of qualified test data and published non-qualified data certainly contributes to lack of understanding and consensus. FOUND project is not targeting to generate design data. Ability to accurately control the applied strain with its rate and measure the cyclic stress response opens new possibilities to clarify the mechanisms involved and develop a mechanism informed model to estimate environmental effects in NPP operation. Great improvement to state of the art in this field is made, if simply fitted trend curves to scattered data sheets can be replaced by a mechanism informed model tuned by accurate data from critical experiments. This is our target. It can be partially achieved already in Safir2018, if our current anticipation of basic mechanisms turns out to be correct.

1.1.4 WP 4 - Fatigue and crack growth caused by thermal loads

Thermal fatigue is an important degradation mechanism in NPP components that can limit the plant lifetime. In thermal fatigue, material is exposed repeatedly to high and low temperatures caused by e.g. plant transients or turbulent flow mixing. Material damage caused by the thermal load cycles can be seen as microstructural changes, residual stresses, material hardening or softening and even cracking after continued loads. The loads cause typically steep through-surface stress gradient with high biaxial stresses on the loaded surface. Exceedence of material yield strength can lead to a local residual stress field. Both experimental [4.1] and computational studies [4.2] have been made of the thermal induced residual stresses and fatigue cracking under known loading conditions. Locations susceptible to such fatigue in NPPs have been e.g. T-joints and piping near leaking valves.
Difficulties in assessing susceptibility and effects of thermal fatigue arise from uncertainties in the load determination and damage initiation and propagation criteria. The multidisciplinary phenomenon requires combination of fluid, structural and fracture mechanics as well as material science fields. For mixing cases, the current numerical assessment methods can be broadly classified as simplified approaches using 1D thermo-mechanical models and/or 1D thermal load, as well as fairly complex 3D approaches using coupling between Computational Fluid Dynamics (CFD) and Finite Element Method (FEM) [4.3-4.5]. As the latter has high computational demand and still requires further validation, the former is often applied in practical analysis work. The simplifications in the 1D approaches lead often to overly conservative analyses or to analyses having unclear validity. Although efforts have been made in the fluid and structural fields for improving the analysis methods, combined CFD-FEM studies of fatigue and crack growth have been limited. Such Fluid-Structure Interaction (FSI) studies are needed to study the thermal fatigue phenomenon through detailed knowledge of the realistic fluctuating thermal and stress fields and to validate and improve the 3D approaches. Analysis of experiments with realistic thermal fatigue damage is needed to improve the damage initiation and propagation criteria.

The project will advance the research by combining the CFD based determination of thermal load with the structural and fracture mechanics based assessment of component fatigue and crack growth. The aim is to quantitatively assess the thermal fatigue degradation by creating a link between the fluid behaviour and material damage. The 1D and 3D approaches are improved and validated by modelling relevant cases with both methods. The initiation criteria of cracks or crack networks with respect to thermal loading are improved by utilizing existing experimental thermal fatigue data together with the FSI and FEM analyses.

1.1.5 WP 5 - Development of RI-ISI methodologies

This work package concerns supporting the planning, realisation, monitoring and supplementation of Finnish risk informed in-service inspection (RI-ISI) programs. The goal of RI-ISI analyses of NPP piping systems is to optimise the inspections so that they are targeted to sites with highest risk. With RI-ISI, the safety of the NPPs can be improved, the irradiation dose to inspection personnel can be decreased, and financial savings can be achieved by removing from the inspection programs sites with small risk but poor approachability. Both European Nuclear Regulatory Working Group (NRWG) and European Network for Inspection and Qualification (ENIQ) have identified several issues concerning RI-ISI that need further research [5.1, 5.2].

For Finnish NPPs RI-ISI is a very topical issue. The regulatory body STUK presently requires RI-ISI analyses for all piping systems concerning both existing NPPs and those under planning/design [5.3]. Both international cooperation and domestic needs indicate a strong motivation for further research. RI-ISI is a multi-disciplinary research topic, combining e.g. risk analysis, structural reliability, fracture mechanics and non-destructive inspection techniques.

1.1.6 WP 6 - Dynamic loading of NPP piping systems

Dynamic loading, especially when connected with other types of loadings, may create critical local stress peaks with a varying amplitude and frequency to pipelines. This may lead to different types of damage or even cause failures. The stress amplitude and frequency are essential factors in the lifetime management. Dynamic loading includes both vibration due to normal operational conditions and sudden very high load peaks due to abnormal processes or accidents. Methods for combining results from different load cases are needed when performing structural integrity analyses. This work package partly aims in determining the best practices and conservatisms of load and result combination methods found in the design standards. The most promising methods of computation of structural responses are studied on the piping structures involving pipe restraints that are case-specifically assumed as linear or nonlinear, respectively. An accordance and recommendation of the selected analysis techniques with the design standards is also reviewed within this WP.

The typical characteristics of design and structural analysis of the NPP piping systems include complex layout geometries, simplified models of actual piping components such as pumps, valve bodies and pipe supports, idealization of piping fluid, loads and required use of result combination methods to evaluate actual physical behaviour. Current topics involving the failure assessment of NPP components include pipelines with different materials, welds and dissimilar metal joints.

For safety reasons, it is important to carry out structural integrity analyses of NPP piping systems accurately and reliably. The reliability of analyses is assured by verification and validation of computational results with reference to the available benchmark experiments and solutions. However, there is a limited amount of test data available especially for complex NPP piping systems and it is a challenging task to precisely define the respective computational models as well. The dynamic behaviour of NPP pipelines has traditionally been analysed with dif-
different FE codes. During the last few years, these FE codes have been linked to different calculation programmes creating a whole pipeline lifetime assessment system covering all types of loads and boundary conditions. For instance, fluid-structure interaction has been solved by linking CFD codes and FE codes. Dynamic structural piping analyses are typically carried out in either time or frequency domain. The main sources of error are due to the linearization, incorporation of the NPP piping components and flowing fluid into the computational model and treatment of both the complex and combined load cases in the frequency and time domains. In the project, the current qualification procedures for structural analyses methods of the NPP piping systems are reviewed in terms of verification and validation with taking into account the known characteristics of the NPP piping systems.

The WP is also partly connected to concrete structures, since pipeline supports are anchored to adjacent concrete structures, which can be damaged by accident loads. The topic of this WP was briefly studied previously already in the FRAS project within SAFIR2010 programme. That study focused on assessing suitability of different element types offered by Abaqus FE code in dynamic structural analyses of pipelines. Special attention was given to the behaviour of primary circuit pipe and one of its restraints in case of pipe break. While offering valuable know-how, that study was however fairly limited and focused strongly on one certain FE code.

1.1.7 WP 7 – Residual stresses in BWR NPPs

Residual stresses play a major role in SCC, which is identified as significant degradation mechanism for various BWR components [7.1]. Experience from ageing NPPs shows 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. Life extension of NPPs requires better understanding of the phenomena that were earlier handled with over-conservative safety factors.

Residual stress relaxation in a BWR NPP was studied recently in co-operation by Aalto University and Teollisuuden Voima Oy in a research project (MACY) funded partly by the National Technology Agency (TEKES). A licentiate thesis [7.2] was written during the MACY project, which was aimed to be the basis for a doctoral thesis in the project. Due to personnel changes, the specific planned thesis cannot be prepared but the research is continued in the project. Residual stress relaxation due to thermal and mechanical loads of two types of mock-up welds were studied by the means of residual stress measurements of as-welded and loaded specimens. Residual stress relaxation and change due to thermal cycling were reported for both specimen types. Ring-core, hole drilling and X-ray diffraction are well known methods for residual stress measurements and were all used in MACY project. Contour-method and FEG-SEM / EBSD, of which the first-mentioned was used in MACY, are methods that can be used for residual stress measurements and developed further in the project to be started.

In this project, samples that were cut from MACY specimens are examined. Real T-joints that were removed from OL1 and OL2 after 28 years of service are also studied. The removal and subsequent examination of these joints presents an internationally unique opportunity to obtain direct measurement data on the residual stress relaxation and prevailing residual stresses in NPP welds.

A repair weld made in the reactor safe-end and its mock-up counterpart provide another unique opportunity to study the residual stresses that form during a repair operation.

1.2 Objectives and expected results

The objectives and expected results of each work package are described in the following subsections.

1.2.1 WP 1 - Remaining lifetime and long term operation of components having defects

The work in 2017 focuses in establishing the relation between the ASME XI [1.1] allowable crack sizes and a FAD based approach. The limits and backgrounds the ASME XI allowable crack sizes are studied and compared with a FAD based approach allowing the estimation of the conservatisms and limits in the applicability. One expected result is a component and load case ASME XI –style representation of allowable crack sizes determined using the FAD based approach.
1.2.2 WP 2 - Susceptibility of BWR RPV internals to degradation mechanisms

This work package provides an investigation on the susceptibility of BWR RPV internals and their supporting structures to various relevant degradation mechanisms. The research work comprises a literature review covering available relevant literature and databases, and a set of computational analyses, including development of new computational applications. The computational part will cover both deterministic and probabilistic approaches. The results of the research work will be documented in a dissertation, as to be completed by the end of SAFIR2018.

The objectives and expected results from this work package concern the following issues:

- Identification of the relevant degradation mechanisms affecting BWR RPV and its internals.
- Collection and documentation of data on the susceptibility of BWR RPV and its internals.
- Computational assessment of the propagation of degradation in the susceptible components of BWR RPV and its internals.
- New computational developments to support assessment of the propagation of degradation.
- Conclusions on degradation potential of BWR RPV and its internals.

As many BWR plants are planning license renewal, it is important to consider more specifically long-term operation (LTO) related structural integrity analyses. These are time dependent degradation potential analyses that cover also the LTO period. The focus of the work in this WP will also be on the LTO phase of the BWR RPV and related components.

1.2.3 WP 3 - Fatigue usage of primary circuit

This work package aims at education of new expert(s) by scientific work, transfer of practical knowhow and international networking for enhanced understanding on accumulation of fatigue usage in NPP primary circuit pressure boundary components. The focus is in “design compatible” experimental approaches to measure cyclic material response and fatigue performance of stainless steels in reactor coolant water and subjected to variable rates of strain during transients. The “design compatibility” ensures direct transferability of laboratory data to fatigue assessment according to design codes: ASME III, RCC-M, KTA and others as well as an unbiased development of RI-ISI programmes. The success rate will be measured through fulfilment of four goals:

- Doctoral thesis of the new researcher is past half way in 2018, beginning as Master’s thesis in 2015.
- FaBello facility provides EU’s best transferable, reliable and accurate tests on environmental fatigue.
- A mechanism informed quantitative model for environmental effects is introduced and experimentally verified to provide improved calculation of F\textsubscript{en} factors – also for transient type strain sequences.
- The transferred and new gained expertise together with the laboratory facility form a sustainable capability to evaluate margins in fatigue design and fatigue usage factors for long term operation of reactors.

1.2.4 WP 4 - Fatigue and crack growth caused by thermal loads

The overall objective in this work package is to combine the CFD based determination of thermal loading with the structural and fracture mechanics based assessment of component fatigue and crack growth. CFD modelling of the thermal load transients for different mixing conditions is validated. The realistic loads are subsequently utilized in structural analyses of the stresses, fatigue life and crack growth. Modelling of fatigue and crack growth with the 1D and 3D approaches enables to validate and improve the assessment methods. In addition, utilizing experimental data of thermal fatigue crack initiation and growth, a link between the thermal load cycles and the material damage is made. The analyses enable more accurate estimation of residual stresses, degradation and strain life for NPP components for thermal fatigue cases. Thermal fatigue in welded components will also be considered.

The work will lead to improved guidance on the use of 1D and 3D approaches to evaluate fatigue and crack growth due to flow mixing based on information obtained from coupled CFD-FEM analyses and comparison with the 1D methods. FEM modelling of the thermal fatigue experiments will lead to more accurate damage initiation and propagation criteria and assessment of crack driving force under thermal loads. In addition to mixing cases, the results can be utilized also more generally, for instance in the assessment of thermal fatigue caused by plant transients. The numerical modelling of the actual flaw under thermal loads provides means to validate simple weight-function based stress intensity factor computation methods common in fracture assessments.
1.2.5 WP 5 - Development of RI-ISI methodologies

The goal of this work package is to support the planning, realisation, monitoring and supplementation of Finnish RI-ISI programs. This work package continues the research work performed in the earlier SAFIR programs. This is realised by providing procedure development and on the other hand applying both that and advanced existing procedures in detailed pilot/benchmark analyses, with which the aim is also to identify the effect of uncertainties in planning risk informed inspection programs. One further aim is to investigate possible mutual support and benefits between probabilistic risk assessment (PRA) and RI-ISI analyses. One starting point for this is that the consequence measure data needed in the RI-ISI analyses is already taken from PRA results. International cooperation is also an important issue. Through WP5, VTT participates in the work of NUGENIA TA8 (formerly ENIQ Task Group Risk) and the EU/NUGENIA research project REDUCE. Topical issues in ENIQ TGR cooperation include research on the connection between RI-ISI and qualification of inspections. REDUCE concerns research on justification of risk reduction of NPP piping components through in-service inspections.

1.2.6 WP 6 - Dynamic loading of NPP piping systems

An aim of this WP is to specify and apply the most applicable computation methods for piping problems in order to provide the means for dynamic analyses of NPP piping subjected to complex load cases, e.g. the water hammer event due to a rapid valve closure.

The work package has three specific goals. The first goal is to carry out a literature review on both the previously developed and most current assessment methods for determination of the loads and structural responses due to water hammer events. The most applicable calculation methods and required input data for piping problems are specified. This extends the 2016 performed survey on suitable piping benchmark problems that can be used for validation and verification of piping analysis software.

The second goal is to develop computation methods for the structural response of piping structures subjected to harmonic steady state and decaying loads due to the water hammer excitation. The computed steady state solution is compared to the corresponding transient response in order to assess a possibility to apply the more computationally efficient numerical procedure to analyse water hammer events on the representative piping structure. In addition, an effect of system nonlinearities due to piping supports on the structural response is assessed.

The last goal is to develop a procedure to combine dynamical spectrum and time-history analyses. Some loads are more or less harmonic by nature while some are more static or on the other hand very dynamic and short. The stress limit criteria (e.g. in ASME III NB-3652) requires that the load cases are combined using the most conservative assumptions if algebraic signs of the load components are missing due to the utilized solution scheme. However, by considering both possible signs, the assessment can be performed more accurately but the amount of possible load combination increases significantly. The goal is to develop a method where the targeted accuracy is maintained while the amount of possible combinations is reduced, e.g. by screening out non-critical time instants and load combinations. Also, an objective fulfilled in 2015 was to develop an evaluation method, with which to replace significant stress cycles in pipe wall that decrease by damping with a smaller number of constant amplitude cycles without any damping.

1.2.7 WP 7 – Residual stresses in BWR NPPs

The main objective of this work package is to gain better knowledge of the residual stresses and their relaxation in pipe welds of a BWR NPP. The expected measurement results will provide unique data from the residual stress state of NPP welds after long service history. The better knowledge and methods for reliable residual stress measurements are gained. In addition, the residual stresses from a repair procedure are measured and compared with previous measurements and used to cross-validate measurements with numerical simulations undertaken in the SAFIR2018 project LOST.

1.3 Exploitation of the results

The main customers are domestic power companies, for which assessment of ageing for management and extension of plant lifetime is very important. Other end users are regulator STUK and EU power companies. FOUND provides new experimental capabilities and more accurate computational lifetime and risk assessment applications for piping systems, BWR RPV internals and other NPP components. Through participation in international networks such as NUGENIA TA8 ENIQ TGR there is valuable international co-operation. The new experimental
and analysis applications developed and tested in the project can later be used in tailored contract works. The planned new developments will provide such new validated services to offer that others do not have.

As for WP1, the evaluation of existing and development of new fracture assessment methods allows performing fracture mechanical assessments more accurately and with increased confidence that is topical when assessing plant lifetime extensions and LTO. The results of this WP can be used in the quantification of uncertainties and estimating confidence in structural integrity assessment of NPP piping systems and components. This includes also the assessment of safety margins.

Concerning WP2, its results can be used for assessing the susceptibility of BWR RPV internals and their supporting structures to various degradation mechanisms. This includes mechanical computational analyses of propagation of degradation. These analysis methods can be used for more accurate lifetime analyses.

Mechanism informed and plastic strain based model for environmental effects is a major goal for WP3. Based on relevant assumptions on underlying mechanisms and accurate in-house data, the model development is expected to lead to a realistic parametric $F_{en}$ model and new state of the art, when compared to the currently used regression fittings and trend curves based on relatively poor scattered data. If successful, this result should be actively defended in international groups and the model proposed for inclusion into international design codes, such as ASME III, JSME and RCC-M.

Studies concerning thermal fatigue in WP4 will provide new and more detailed information on loads caused by flow mixing and material response to related cyclic thermal loads. The methods developed and validated in the work can be applied for more accurate lifetime analyses with reduced conservatism.

As for WP5, its results can be used for planning, realisation, monitoring and supplementation of Finnish RI-ISI programs. The results of WP5 can also be used outside RI-ISI programs, in various reliability analyses and NDT inspection reliability assessments.

The results of WP6 can be used for more accurately assessing both the dynamic behaviour of piping systems and stresses experienced by restraining pipe supports. An evaluation of the load combination methods allows reducing conservatisms in results. The research in developing best practices for qualifying piping analyses provides valuable information for validation purposes of existing and new analyses and software. These analysis methods can be used for more accurate and reliable lifetime analyses.

For WP7, results can be used by NPP operators for more accurate lifetime analyses with reduced conservatism in connection to life extension of the existing NPPs and the design life of the new ones.

1.4 Appropriateness of the project to SAFIR2018 programme

Concerning SAFIR2018 Framework Plan [8.1], the topics of the FOUND project are strongly connected to Section 3.4 Structural safety and materials, and therein especially to Sections 3.4.4.1 Assessment of failure and damage mechanisms, 3.4.4.3 Non-destructive examination and assessment methods, 3.4.4.4 Advanced assessment methods for structural safety, 3.4.4.5 Life cycle management methods and life cycle extension, and 3.4.4.7 Interface between probabilistic and deterministic design.

Research subjects related to the load follow mode in NPP operation mentioned specifically as an important research topic in the update to the Framework plan for the 2017 SAFIR call [8.2] can be identified in most of the planned work packages. Load follow typically leads to cyclic changes in the mechanical load at some NPP components which can accelerate the ageing processes, increase risk and influence the lifetime expectancy of the components and plant in general. The planned work packages of this project are related to the component lifetime and ageing assessments (WP1, WP2, WP5) and evaluate directly material and component performance under cyclic loading (WP3, WP4, WP6). Also, cyclic loading has a tendency to influence the residual stress state of the component that is evaluated in WP7.

1.5 Education of experts

Young scientists will be working in the project with problems related to nuclear components. Knowledge transfer from experienced scientists to recently graduated scientists will continue in the project. In WP2, a doctoral thesis will be prepared concerning degradation mechanisms of RPV internals. In WP3, the work started with a master’s thesis in 2015 concerning transferability of laboratory data to real-life components is continued targeting to a PhD on mechanism informed model on environmental effects in fatigue. In WP4, a licentiate’s thesis will be prepared concerning thermal mixing. In WP6, a Master’s Thesis was prepared on linearization of piping supports in 2015 [6.1].
1.6 International co-operation

Within the project FOUND, VTT researchers participate in the activities of the international NUGENIA association that aims to foster collaboration between the nuclear industry, research organizations and technical safety organizations. The most important participation in the NUGENIA work is in TA8 focusing on in-service inspection and NDE. Through this platform, VTT has benchmarked and advanced its RI-ISI methodology partially developed in SAFIR/FOUND and in the international REDUCE project. In addition, the PRA and RI-ISI experts involved also in FOUND are contributing to the preparation of a new H2020 EU project proposal on reducing the number of RI-ISI inspections. If successful in the application, the RI-ISI work in FOUND can benefit from the research performed in the EU project. Another international work group where VTT researches participate through FOUND is the ETSON EG2 concentrated on RPV integrity aspects. The objective of the expect group activities is to improve the mutual understanding of the different approaches used in the RPV integrity assessment in different countries and the identification of differences between the approaches as well as possible gaps in the regulations.

The research, methods and results obtained in the FOUND project facilitate participation to international benchmarks such as those organized by the OECD NEA CSNI Working Group on Integrity and Ageing of Components and Structures (WGIAGE). The participation in the upcoming WGIAGE LBB benchmark, where the methods developed in the RI-ISI related work can be utilized, is considered.

An important international co-operation is with the BG Group that consists of plant engineers from Swedish and Finnish nuclear power companies. The BG Group has provided funding for the FOUND project for aspects relating the integrity and ageing assessments of BWR components and the project work benefits greatly from the insight of the expert engineers working daily with the integrity assessments of the actual plants.

The environmental fatigue research in WP3 is strongly visible and connected internationally. An agreement over relevant nuclear-grade stainless steel has been reached with a NPP vendor. The material will be used in the low-cycle environmental fatigue experiments. Co-operation with the international material supplier ensures that the required background information of the test material is obtained thus reducing the need for VTT to perform repetitions of the typical characterization and fatigue testing for the material. The results of the environmental fatigue testing are actively published in the ASME PVP conference that also gathers the experts in the field of fatigue engineering.

The studies of the welding residual stresses and their relaxation in WP7 are connected to the planned European RESTRESS and GEMMA projects where both the comparative measurements and comparison of the measured and computed residual stresses are carried out. The measurement techniques developed in FOUND can be benchmarked and the obtained results compared within these international projects.
2. Work plan

The general work plan for 2015-2018 and detailed plan for project year 2017 for each work package is presented in the following. The detailed budget of the project is presented in the Annexes and is only summarized for each work package here.

The original scope in WP1 was related to the EU project ATLAS. For 2016, the work package was re-aimed to consider and solve issues directly related to component integrity evaluation methods and LTO topics.

2.1 WP1 – Remaining lifetime and long term operation of components having defects

The work package 1 for 2017 focuses on methods used for evaluating remaining the lifetime and LTO of components with defects. The work in the topic was started in 2016 when a post-processing method for evaluating J-integral in thermal and residual stress fields was developed. The method allows also evaluation of the J-integral for non-proportional loading cases by estimating the strain energy density from the stress field. The developed method still needs to be tuned and validated but this work is postponed to 2018.

Partners and person months allocated to WP1 for project year 2017 are presented in table below. The total volume of the work package is 12 k€ which is allocated to the one performed task. The work package is funded by VYR (8 k€) and VTT (4 k€).

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2.1.1 Evaluation of allowable crack sizes (T1.1)

The task evaluates the maximum allowable crack sizes defined in ASME XI [1.1] in piping components. The allowable crack size limits given in IWB-3500 depend only on component classification, nominal wall thickness and the crack aspect ratio. Cracks in piping components exceeding these limits are allowed to be analysed individually using for example the flaw evaluation methods given in ASME XI Appendix C and Appendix H. Appendix H is a FAD-based procedure combining all typical mechanical failure modes whereas Appendix C gives individual methods for determining the proximity to plastic collapse, ductile crack growth and brittle fracture separately. The aim of the work is to evaluate the Appendix C and Appendix H methods against a FAD-based FFS-method. The British Standard BS7910:2013 [1.2] that has been developed independently of the American ASME rules will be applied for the comparison.

A limited survey on the rationale and built-in safety factors behind the ASME XI methods will be performed. The applicability of these methods will be discussed; for which materials, loadings and crack growth mechanisms are they applicable to. Computational examination of example cases allows the comparison between the ASME XI and BS7910 methods and the evaluation possible conservatisms in the ASME XI methods.

In addition, the FAD based evaluation method will be converted to such a form that plots the allowable crack depth and length for the given load case based on the condition that all the points lie on the FAD. The developed approach can be utilized to determine the sizes of the critical defects for the component and later extended to evaluate the remaining lifetime required in LTO assessments.
2.1.2 ETSON Expect Group on Mechanical Systems (EG2)

This task is not included in the project due to budget cuts.

ETSON, the European Technical Safety Organisation Network, is the European association of nuclear assessment bodies that perform safety assessments in support of their national authorities. The collaboration in ETSON aims to contribute to the harmonisation of nuclear safety practices in Europe. The ETSON EG2 working group is concentrated on Reactor Pressure Vessel (RPV) integrity. The objective of the group activities is to improve the mutual understanding of the different approaches used in the RPV integrity assessment in different countries and the identification of differences between the approaches as well as possible gaps in the regulations. The resources in this task are dedicated for participating in the work of the ETSON EG2.

2.2 WP2 - Susceptibility of BWR RPV internals to degradation mechanisms

This work package provides an investigation on the susceptibility of BWR RPV and its internals to various relevant degradation mechanisms. The work consists of a literature review, covering available relevant literature and databases, and of a set of computational analyses, including development of new computational applications. The computational part covers both deterministic and probabilistic approaches.

The main purpose is to prepare a dissertation based on research work of only one researcher. Due to this, the work package is not divided further into tasks.

The following actions describe the work plan for 2015-2018:

- To identify the relevant degradation mechanisms affecting the BWR RPV internals and their supporting structures.
- To collect and document data on the susceptibility of the BWR RPV internals to degradation.
- To computationally assess the propagation of degradation in the susceptible BWR RPV internals.
- To provide new computational developments for assessment of the propagation of degradation.
- To provide conclusions on degradation potential of BWR RPV internals.

The first two items were carried out in 2015. Most of the third and fourth items were carried out in 2016. In 2017, the third and fourth items will be extended and completed, together with doing the fifth item.

As many BWR plants are planning license renewal, it is important to consider more specifically long-term operation (LTO) related structural integrity analyses. These are time dependent degradation potential analyses that cover also the LTO period. Thus, in addition to what is described above, the specific planned work topics for 2017 include:

- Degradation potential analyses to:
  - Core shroud support legs,
  - Water level measurement nozzles,
  - Steam separator support legs, and
  - Spring beam brackets.
- Implementation, testing and validation of the more straightforward probabilistic crack growth computation procedure developed in 2016, see below for a brief description of the procedure.

The developed probabilistic crack growth computation procedure concerns interpolation between a large enough representative set of case specifically computed crack growth simulations. These simulations cover all input data parameters considered as probabilistically distributed. In terms of each of such parameters, it is sufficient to compute a limited number of crack growth simulations, covering the range of validity of the given underlying distributions. Then, in the probabilistic crack growth analysis, the crack growth history is obtained for each realisation by interpolating between those two simulated curves between which its randomly picked initial crack size resides. The obtained results are used for degradation state probability computations with the Markov application, up to the probability of failure, corresponding e.g. to pipe leak. Compared to any existing Monte Carlo procedure, involving at least several thousands of crack growth simulations applying fracture mechanics, the needed computational work using simple interpolation is much smaller.

This work involves both co-operation with and funding from TVO. Partners and person months allocated to WP2 for project year 2017 are presented in table below. TVO is listed in the table as co-operation is essential in
this task but no concrete funding will be allocated for their work. The total volume of the work package is 41 k€. The work package is funded by TVO (18 k€), Swedish-Finnish BG-Group (15 k€), VYR (3 k€) and VTT (5 k€). The dissertation manuscript is expected to reach the finalization phase during 2017 and the aim is also to apply for 2 person months of VTT funding for the completion.

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2.3 WP3 - Fatigue usage of primary circuit

This work package continues the work started with a Master’s thesis in 2015 aiming to gain practical knowhow and learn about international progress and challenges related to transferability of laboratory fatigue data to primary circuit fatigue assessment and usage monitoring. This will be supported with a couple of experiments using the FaBello facility in hot and pressurised reactor coolant water. The Master’s thesis addressed the confusing roles of test control mode and temperature in $F_{en}$ calculation. The US and Japanese calculation models [3.2,3.11,3.12] assume no effect of temperature without environment. Recent results both with stabilised and non-stabilised grades show that this is an incorrect assumption for low strain amplitudes and VTT data reveals that the test temperature and strain rate may be responsible for up to half of the measured “environmental” effect [3.7,3.8].

This will introduce a path to longer term research (PhD) on stainless steel complex cyclic response and fatigue performance in reactor coolant water. By the end of FOUND project in 2018, the main objective of WP3 is to reveal the underlying mechanisms and develop a model to quantify effects of hot water environment in fatigue of stainless steel. A new model for $F_{en}$ calculation will be proposed in the PhD, which may be completed by 2018 or soon after. Plastic strain is more relevant parameter than total strain for fatigue, and most probably this holds also for environmental effects. The PhD work will begin with a hypothesis that $F_{en} = f(\sigma_p)$, instead of $F_{en} = f(\sigma_{tot})$, which was assumed through regression fitting of strain life data in Japanese and US data banks. The new plastic strain ($\sigma_p$) based model shall be applicable to scientific research and measuring of parametric influences compatible with design methods for NPP components (in contrast to NUREG/CR-6909, which is not fully compatible with ASME Section III). A particular task is to explain the lower $F_{en}$ data reported for variable strain rate tests, e.g. by the "Areva EPR transient". Experiments to separate environmental effect from other effects and adjusting elastic and plastic strain rates to simulate transient type straining will continue in 2017-2018.

In 2016, an agreement with an international partner was reached over the supply of AISI 304L stainless steel for the environmental fatigue tests. The use of a relevant NPP grade piping material is essential to increase the transferability of the test results to the actual plant conditions. First tests with the material are performed in late 2016 and the material will be utilized in the subsequent project years 2017 and 2018.

Partners and person months allocated to WP3 for project year 2017 are presented in the table below. The total volume of the work package is 98 k€, which is allocated to the two tasks as 72 k€ and 26 k€, respectively. The work package is funded by VYR (65 k€) and VTT (33 k€).

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<th>Partners in WP3</th>
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The main work is allocated for young scientist(s). VTT senior staff role is to perform the special skills and safety concerned autoclave experiments and to supervise the research. The following tasks are planned in 2017.

2.3.1 Experimental Fatigue R&D in LWR Coolant Water (T3.1)

- Four fatigue tests in PWR environment will be designed and performed using direct strain control. The test parameters will be based on the 2015 and 2016 results to provide critical data for the model development (see Section 2.3.2).
- The plan is to utilize lower or higher strain amplitude than in the tests performed in 2016. The lower strain amplitude is preferable to emphasize the effects of the loading transient type on the fatigue life but the test duration and costs are increased with lower strain amplitudes. The selection of the strain amplitude is made when the results from the 2016 tests are finalized.
• Remaining budget after the FaBello tests will be used to perform strain-controlled fatigue tests in air for improved characterisation of the test material.

2.3.2 Mechanism based model to justify revised Fen (T3.2)

• The results from 2015 and 2016 on environmental effects and transferability of laboratory fatigue data to assessment of primary piping components have been considered in the development of the Fen model. The newer results obtained in the 2017 test campaign will give more feedback on the hypothesis that Fen = f(ε), instead of Fen = f(εtot) and will be used to further develop and validate the model. The results will also affect the planning of the subsequent 2018 test campaign.
• Progress in developing the new Fen model will be published at a relevant conference (e.g. ASME PVP 2017). This may also be complemented with a scientific journal paper.

2.4 WP4 - Fatigue and crack growth caused by thermal loads

Components experiencing turbulent mixing susceptible to thermal fatigue are modelled with CFD by using the Large-Eddy Simulation (LES) turbulence model. Validity and accuracy of wall functions, which are usually required in real-life mixing CFD calculations, are assessed. The intensity and frequency content of the thermal fluctuations in the different mixing situations is also studied. In particular, it is studied how approximate scaling laws for the intensity and frequency content, developed in the previous FRESH project, apply in the different cases.

The 3D transient thermal loads are imported into FEM for structural analyses. Fatigue and crack growth calculations by using 1D and 3D thermo-mechanical models are carried out. Of particular interest are the so-called large-scale fluid motions, which typically have relatively low frequency, leading potentially to deeper cracks. Another consequence of the low frequency is that the 1D model may become non-conservative in certain cases, and therefore different boundary conditions of the 1D model are studied. The realistic 3D thermal loads are utilized as more accurate input parameters in the 1D methods, and uncertainties in the 1D assumptions are reduced.

Thermal fatigue modelling capabilities are developed by numerical modelling of thermal fatigue experiments. Utilizing experimental data together with numerical analyses, initiation and propagation of individual cracks and crack networks will be studied. Thermal fatigue cracks formed due to repeated loading will be included in the analyses to determine their effect on the high biaxial near-surface stress state and steep gradient typical for thermal loading. Also the effects of load history and material strain hardening properties on crack initiation stresses are estimated. Combining experimentally measured thermal load cycles and determined cycles-to-failure data of fatigue crack growth with the numerical results of crack driving forces, a strain-life curve for thermal fatigue can be estimated. The developed and utilized method allows estimation of crack interaction, networking and possible shielding effects.

Partners and person months allocated to WP4 for project year 2017 are presented in table below and the work plan for 2017 is described in the following sub-sections. The total volume of the work package is 36 k€, which is allocated to the two tasks as 18 k€ and 18 k€, respectively. The work package is funded by VYR (22 k€) and VTT (14 k€).

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2.4.1 Coupled CFD-FEM modelling of thermal fatigue (T4.1)

In 2015-2016, crack growth due to turbulent mixing of hot and cold water was studied for the FATHER experiment and for a feedwater mixer of a BWR. For the FATHER experiment, thermal mixing loads from transient CFD simulations were utilized. For the feedwater mixer, the spectrum method was further developed and used for having computationally inexpensive but realistic thermal mixing loads. In both cases, crack growth results from using the simplified but conservative sinusoidal (SIN) method were compared with the results from using the more realistic loads. In 2017, corresponding fatigue calculations are performed by utilizing the CFD and spectrum loads from the previous project years. In case of the FATHER experiment, the fatigue times obtained by using the CFD, spectrum and SIN loads are compared with each other and with the experimental data. For the feedwater mixer, the fatigue times obtained by using the spectrum and SIN loads are compared. In both cases, the fatigue and
crack growth times are compared with each other. A journal article is written from the fatigue and crack growth results from using the different thermal mixing load definitions.

2.4.2 Evaluation of low-cycle thermal fatigue cracking (T4.2)

The work continues the work performed in the numerical modelling of low-cycle thermal fatigue. During the two previous project years the focus has been in including the actual flaw in the computations and assessing the thermal fatigue crack growth path and rate based on the computed crack tip opening displacement. This approach has been shown to improve the estimation considerably over the typical approach of utilizing the uncracked stresses and tabulated handbook stress intensity factor solutions. This is due to the considerable plasticity typical to low-cycle thermal fatigue and the limited applicability of the linear fracture mechanical methods to such cases. In the proposed approach, the crack driving force is determined from the crack tip opening displacement. The crack driving force and cycles to failure results obtained in 2015 corresponded far better to the experimental data than to the results obtained previously with uncracked samples. The work has showed the importance of the cyclic material response on the crack loading and growth rate. In 2016, an effort was made to better justify the approach and compare the predicted growth rates with experimental data.

It is not feasible to suggest that complex crack opening simulations are required to assess the susceptibility to thermal fatigue. Therefore, in 2016, the conditions that are experimentally known to induce thermal cracking were evaluated and the critical stress and strain conditions prone to induce cracking were identified. A reliable method for assessing crack growth rates using the stress distributions is yet to be established.

In 2017, the work will focus on two main topics: justifying the use of the CTOD as a crack driving force parameter for thermal fatigue and the development of a crack growth rate estimation method based on the uncracked stress and strain state in the thermally cycled specimen. A conference paper will be prepared on the CTOD justification.

2.5 WP5 - Development of RI-ISI methodologies

This work package provides further development of RI-ISI analysis procedures. These developments and improvements will be tested in detailed pilot/benchmark analyses, mostly by using the quantitative VTT RI-ISI analysis procedure. In addition, connection between PRA and RI-ISI analyses is investigated. The developments for 2017 include improving and expanding the computational application of the VTT RI-ISI analysis procedure as well as possible co-operation with SAFIR2018 project WANDA. As for international co-operation through this work package, VTT participates in the work of NUGENIA TA8 (formerly ENIQ TGR).

This work involves both co-operation with and funding from TVO.

Partners and person months allocated to WP5 for project year 2017 are presented in the table below and work plan for 2017 is described in the following sub-sections. TVO is listed in the table as co-operation is essential in this task but no concrete funding will be allocated for their work. The total volume of the work package is 35 k€, which is allocated to the three tasks as 13 k€, 11 k€ and 11 k€, respectively. The work package is funded by TVO (6 k€), VYR (17 k€) and VTT (12 k€).

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2.5.1 Probabilistic LBB assessment (T5.1)

This task is left out of the project due to budget cuts.

In the current VTT RI-ISI procedure, the pipe break probability is evaluated from the state where the analysed flaw has penetrated the outer surface of the pipe. This state can be more accurately classified as leak given that the LBB conditions are fulfilled. A significant margin may still exist between the leak state and break state that occurs when the inside surface crack propagated through wall becomes unstable.
The goal of this task is to survey the applicable and benchmarked LBB procedures in the literature (such as US Standard Review Plan (SRP) 3.6.3, “Leak-Before-Break Evaluation Procedures” or the German break preclusion (BP) concept) and to implement the best-suited approach to the VTT RI-ISI toolkit. Particularly, the work includes selecting suitable flaw re-characterization techniques to transform the inside-surface crack into a through-wall crack, implementing the stress intensity factor solutions for the through-wall crack configurations and the leak rate calculations for the through-wall cracks.

It is expected that the inclusion of the leak and break criteria in the assessment gives an additional confidence and margins over the current procedure that pessimistically classifies all internal cracks grown near the external surface as failed. The margin against failure can also be increased for the cases where LBB cannot be argued if the material fracture initiation toughness can be considered instead of the crack depth alone.

The implemented LBB method can be validated in the upcoming international OECD LBB benchmark where the aims are to evaluate the country-specific approved LBB guidance and to determine the differences among the different LBB procedures. A full benchmark participation cannot be planned due to the estimated resource demand that far exceeds a feasible amount for this task.

2.5.2 Extension and update of quantitative VTT RI-ISI analysis procedure (T5.2)

The purpose of the work is to develop further the quantitative VTT RI-ISI analysis procedure for NPP piping components. This will be realised by both improving the computational efficiency and extending its capabilities. The existing VTT RI-ISI procedure consists of two modules, one for crack growth simulations, and the other for leak probability and risk computations. The intention is both to combine these modules and to improve the efficiency of the leak probability computations that are based on the Markov and Monte Carlo methods. The work includes also example analyses for representative pipe components as well as reporting the achieved results.

The capabilities of the VTT RI-ISI analysis procedure are extended by incorporating the Monte Carlo method. In probabilistic crack growth simulations, the values for certain input data parameters are taken at random from the respective distributions, and the stress intensity factor values for a range of the initial crack dimensions remain very small with regard to a total number of simulation cases involved. These cases are screened out from further computations by using applicable crack propagation threshold data.

The possibility for screening the assessment cases during the computation is included to improve the efficiency of the simulations. The screening rules are determined based on the attained time periods of the set of cracks to grow through-wall. The screening procedure significantly reduces the computational cost of the Monte Carlo method. An applicability of screening is also considered for the Markov process. New results, i.e. the critical crack dimensions that affect the leak probability, are from now on included in RI-ISI reporting.

Another development of VTT RI-ISI concerns probabilistic distributions for material and environment specific parameters that are used in modelling SCC with the disposition curve equations. Such distributions were derived in 2016 in cooperation with the SAFIR2018 project THELMA. In 2017, the intention is to incorporate those distributions into the VTT RI-ISI procedure. In addition, the possibility of including new probability of detection (POD) functions into VTT RI-ISI is assessed in cooperation with SAFIR2018 project WANDA. These POD functions will be pipe component and degradation mechanism specific. They are planned to be created within WANDA. If the PODs are feasible for RI-ISI analyses, they will be implemented to the VTT RI-ISI toolkit in this work package.

2.5.3 Connection between PRA and RI-ISI analyses (T5.3)

The general aim of the task is to study the possible mutual support and benefits between PRA and RI-ISI analyses. The current situation is that the consequence measures of piping failures used in RI-ISI analyses are calculated using PRA model. However, in the literature study performed in 2015, it was found that PRA software could support the consequence measure computation better. Some other development possibilities were also identified, but improvements to the consequence measure computation were considered most promising.

In 2016, a new method was developed to improve the computation of the piping failure consequences from PRA model and it was demonstrated using FinPSA software. Using the method, the conditional core damage probabilities (CCDPS) of all piping component failures can be calculated automatically at once in the same table based on the results of the PRA model. The method enables detailed modelling of the consequences of piping component failures without complicating the PRA model itself.

Conditional large early release probability (CLERP) is another important consequence measure, and formulas for its computation were also outlined in 2016. For 2017, the plan is to study the computation of the CLERP more comprehensively and add it to the method developed in 2016. The computation of the CLERP is much more chal-
lenging than the computation of the CCDP, because it requires integrated level 1 and level 2 PRA computations which use different methods.

2.5.4 NUGENIA TA8 (formerly ENIQ Task Group Risk) activities (T5.4)

The following description concerns mainly the work plan for 2017. VTT’s participation to NUGENIA Technical Area 8 (TA8): “In Service Inspection, Inspection Qualification and NDE Evaluation” Sub-Area Risk (SAR) (formerly ENIQ TGR) meetings and activities is provided through this work package. This concerns participation in the work of TA8 in developing recommended practices and discussion documents related to RI-ISI. The following topics are under development within TA8 SAR:

- Risk informed pre-service inspection (RI-PSI) for new plants,
- RI-ISI for long term operation,
- The scope of RI-ISI for a complete NPP,
- Planning of continuation project for REDUCE
- Reviews of RP9 “Verification and Validation of Structural Reliability Models and associated Software to be used in Risk-Informed In-Service Inspection Programs” and RP11 “Guidance on Expert Panels in RI-ISI”.

Of considerable interest is to follow the development in the first three topics. Concerning the third topic, VTT participated in the REDUCE project, which was completed in September 2016. The other topics are very interesting too. During 2017, new issues can be added to the list of topics to be developed.

2.6 WP6 - Dynamic loading of NPP piping systems

This work package provides more accurate assessment of both the dynamic behaviour of piping systems and the stresses experienced by restraining pipe supports. In addition, the work focuses on engineering methods combining dynamical spectrum and time-history analyses as well as the effect of free vibration damping on significant stress cycles affecting component lifetime. These analysis methods can be used for more accurate lifetime analyses. These developments and improvements will be tested in detailed pilot/benchmark analyses.

In 2016, the procedure of linearization of piping supports was applied to a representative model involving non-linear supports with gaps and frictional contacts. The solutions on test models did not, however, result in sufficient reliability and accuracy level required for NPP applications. The work plan for 2016 also included the literature review on the previously developed methods for qualification of dynamical analyses of large-scale piping systems. However, the review did not include any benchmark problems for qualification of structural analyses involving water hammer loads. The accurate structural response is required for the computation of the stress intensity factors and fatigue crack growth rate for the piping components subject to water hammer loads. For 2017, the important water hammer topic covering both a literature review and development of computational methods will be introduced.

This work involves both co-operation with and funding from TVO.

Partners and person months allocated to WP6 for project year 2017 are presented in the table below and work plan for 2017 is described in the following sub-sections. TVO and FEMdata are listed in the table, as co-operation is essential in this task but no concrete funding will be allocated for their work. The total volume of the work package is 34 k€, which is allocated to the three tasks as 7 k€, 10 k€ and 16 k€, respectively. The work package is funded by TVO (9 k€), VYR (12 k€) and VTT (12 k€).

<table>
<thead>
<tr>
<th>Partners in WP6</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>2.2</td>
</tr>
<tr>
<td>TVO</td>
<td>in-kind</td>
</tr>
<tr>
<td>FEMdata</td>
<td>in-kind</td>
</tr>
</tbody>
</table>
2.6.1 A literature review on the methods for water hammer analyses (T6.1)

The work plan for 2017 includes the literature review on both the previously developed and most current assessment methods for determination of the loads and structural responses due to water hammer events such as a rapid valve closure. The review involves an assessment of accordance of these techniques with both the EPRI Handbook [6.2] and design standards by the American Society of Mechanical Engineers (ASME) [6.3] and by the American Institute of Steel Construction AISC [6.4]. The most applicable calculation methods and required input data for representative problems are specified in order to provide the means for water hammer analyses of the most typical cases of NPP piping. The most promising methods of estimation of the displacements and reaction forces are studied on the representative piping structures in the related work package T6.3.

In summary, the review is carried out in terms of the water hammer assessment procedures with taking into account the treatment of nonlinearity, idealisation of the NPP piping components and flowing fluids and accordance of the methods with the standards [6.2,6.3]. An assessment of the more detailed water hammer models and numerical methods are considered in T6.3.

2.6.2 Combination of moments from dynamic analyses (T6.2)

Combination of pipe moments from different simultaneous dynamic time history analyses have been studied in the previous project years. Methods for combining the moments in order to calculate the resultant moment have been studied analytically and numerically. In the numerical studies, moment time signals generated randomly or from FPIPE analyses have been used. More effective combination methods were proposed, but taking the algebraic signs of the moments into account is difficult.

In 2017, the procedure developed by TVO for combining forces and moments at piping and component supports is applied and developed for the pipe moments. The new procedure allows taking the algebraic signs of the moments into account. The moment combination procedure is tested, validated and documented by using moment time histories obtained from dynamic FPIPE analyses. The procedure consists of i) dividing each signal into the positive and negative parts so that the algebraic signs can be accounted for, ii) removing from each dynamic load case time points for which there exists other time points having larger moments and iii) searching all possible time point combinations between different load cases for the remaining values. The possibility of removing further time points by using suitable criteria is also studied.

2.6.3 A development of computation methods for the water hammer excited response of piping structures (T6.3)

In 2017, the development of computation methods for the water hammer excited response of representative piping structures will be carried out. For computational efficiency, the modal dynamic procedure is used to determine the harmonic response of a linear system that is modelled with pipe elements. For decaying loads, the transient modal dynamic analysis is utilized to obtain the response of the linear system to general loads. The general solution of the wave differential equation can be utilized to model the transient water hammer excitation due to a rapid valve closure event. The most applicable solutions of T6.1 for estimation of structural forces due to water hammer events are adopted.

The transient dynamic analyses are carried out both with the linear solution methods for computational efficiency and nonlinearly to take into account the possible system nonlinearities e.g. due to piping support behaviour. The steady state solution is compared to the transient response in order to assess a possibility to apply the more computationally efficient numerical procedure to analyse water hammer events on the representative piping structure. The comparison also allows an estimation of the nonlinear effects on the water hammer response. The more complex piping configurations with related components such as pumps are, however, considered as a future subject.

2.7 WP7 - Residual stresses in BWR NPPs

Residual stress relaxation in BWR NPP was studied with co-operation of Aalto University and Teollisuuden Voima Oyj in FIMECC Demanding Applications programme (MACY project). The results are presented in Miikka Aalto’s Licentiate’s thesis, which was completed in 2015. Residual stress relaxation due to thermal and mechanical loads of two types of mock-up welds were studied by the means of residual stress measurements of as-welded and loaded specimens. Residual stress relaxation and change due to thermal cycling were reported in
both specimen types. Ring-core, hole drilling and X-ray diffraction are well known methods for residual stress measurements and were all used in MACY project. Contour-method and FEG-SEM / EBSD, of which the first-mentioned was used in MACY, are methods that can be used for residual stress measurements and are developed further in this project.

Partners and person months allocated to WP7 for project year 2017 are presented in table below. The total volume of the work package is 64 k€. The work package is funded by VYR (45 k€) and Aalto University (19 k€).

<table>
<thead>
<tr>
<th>Partners in WP7</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>Aalto University</td>
<td>5.5</td>
</tr>
<tr>
<td>TVO</td>
<td>in-kind</td>
</tr>
</tbody>
</table>

The first main focus of the work package is the examination of samples that were cut from MACY specimens and the examination of real T-joints that were removed from OL1 / OL2 after 28 years of service.

In 2015, FEG-SEM EBSD mapping was used to measure the amount of residual plastic strain in the previously loaded laboratory welds and calibration samples. During 2016, this analysis is continued, and the measured values are compared with information from the removed T-joints. It is unclear, whether EBSD measurement from actual removed samples will be available. Even in the absence of EBSD samples from actual components, the analysis of laboratory samples provides important supporting data for understanding the development of residual stresses in the actual components.

Also in 2015, the T-joints sections were removed from OL1 and OL2 plants. The decontamination of these components is on-going at the time of preparing this application. Therefore, the research work during 2015 focused on development of residual stress measurement techniques and preparation of the actual measurements on the removed components. Significant development of the contour method to produce residual stress maps of a specimen cross-section was achieved.

In late 2016, the removed T-joints are measured with the prepared measurement techniques. This presents a unique opportunity to evaluate actual in-service residual stresses after long-term service. However, it also presents number of unique challenges, including working with decontaminated components with some possible residual contamination. Consequently, the work includes use of several residual stress measurement methods and comparison with laboratory samples to prepare for possible limitations during actual measurement.

The residual stress related research at the second phase of the project focuses on the examination of the feedwater nozzle residual stress state due to repair welding. The work in 2017 focuses on the measurements of the residual stress state due to repair welding. A full mock-up sample of the repair as well as additional samples made for welding validation purpose are examined in the work as described below.

2.7.1 Residual stresses due to repair welding (T7.1)

In 2017, a welding repair procedure is completed in for a BWR RPV nozzle safe-end. A full-size mock-up of the repair is done for validation of the repair welding procedure. This mock-up can be measured with non-destructive means (X-ray diffraction) before and after the repair welding trials. Additional samples are made for weld procedure validation. These (if available) are measured using destructive means (contour method). The measured residual stresses will be compared with stresses previous measurements and with simulation results obtained in the SAFIR2018 project LOST. The residual stress state due to repair welding for a similar mock-up is modelled in LOST and thus the measurements offer direct comparison and cross-validation with the associated modelling effort. Should the said mock-up not be available for measurements, additional measurements on available extracted welds and available samples are completed to compensate.

2.7.2 FEM studies of the plasticity produced by thermal cycles (T7.2)

This task is not performed in 2017 due to budget cuts.

Residual stresses and their relaxation in BWR pipe weld mock-ups were experimentally studied in the MACY project. Thermal cycling was observed to change the residual stresses mostly in the surface layer and to a certain depth inside the material. FEM studies are needed to provide the relationship of the relaxation depth and the
strain history of a thermal cycle. Means of this research are to compare FEM studies of plasticity to the measured residual stress changes in the previously studied samples.

In 2015, FEM was used to explain measured residual stress values in laboratory samples as well as to help combine different measurement techniques (with different uncertainties) into a unified description of weld residual stresses and their relaxation during thermal loading. The task was put on hold for 2016.

In 2017, additional FEM studies are completed to correlate measurements made from the thermally loaded sections on the pipe sections with previous measurements and expected loads.

### 2.8 WP8 - Project management

A separate work package is dedicated to project management for budgeting and result evaluation purposes. All project management related work and costs will be allocated to this WP. For 2017, the estimated resource allocation for project management and result review is 0.8 person months. The total volume for project management and review work package is 13 k€ which is funded by VYR (8 k€) and VTT (5 k€).

The need for separate evaluation on how the obtained results can be exploited by the end users has emerged in the discussions with project partners and members of the reference group. Therefore, a short summary that evaluates the utilization of the results obtained in the previous project year will be prepared in this work package. The aim of the evaluation is to determine the possible practical use and implementation of the results and provide first-hand guidance on the subject.
### 3. Deliverables 2017

The planned deliverables for 2017 are listed in the table below.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>VTT research report on limits and backgrounds the ASME allowable crack sizes and comparison with a FAD based method</td>
<td>0.8</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Full dissertation manuscript: Susceptibility of BWR RPV internals to degradation mechanisms.</td>
<td>1.4</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D2.1.2</td>
<td>VTT research report: degradation potential analyses to BWR RPV internals and application of developed probabilistic crack growth computation procedure.</td>
<td>1.4</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Presentation and slide set on the results of fatigue experiments in air and PWR water.</td>
<td>2.2</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.1.2</td>
<td>Abstract and draft paper submitted to PVP2018 or some other conference on experimental methods for environmental effects and/or test results.</td>
<td>1.0</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D3.2.2</td>
<td>Updated research plan for studying the mechanisms and developing a new $F_{cr}$ model.</td>
<td>1.6</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D4.1.1</td>
<td>A journal article manuscript on the fatigue and crack growth rates resulting from different thermal mixing load definitions</td>
<td>1.1</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D4.2.1</td>
<td>Conference paper on the use of CTOD as a crack driving force parameter for thermal fatigue</td>
<td>1.0</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D5.1.1</td>
<td>VTT research report on the implementation of LBB methods to VTT RI-ISI procedure Removed due to budget cuts.</td>
<td>0.0</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D5.2.1</td>
<td>VTT research report on VTT RI-ISI procedure extended to in-</td>
<td>0.9</td>
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<tr>
<td></td>
<td>Project title</td>
<td>PM</td>
<td>Date</td>
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<tr>
<td>D5.3.1</td>
<td>VTT research report on the computation of CLERP within the PRA software</td>
<td>0.7</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D6.1.1</td>
<td>VTT Research report on a literature review on the computational methods for water hammer response analyses</td>
<td>0.5</td>
<td>30.9.2017</td>
</tr>
<tr>
<td>D6.2.1</td>
<td>VTT Research report on the testing and validation of the developed moment combination procedure.</td>
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<td>31.12.2017</td>
</tr>
<tr>
<td>D6.3.1</td>
<td>VTT Research report on a development of computation methods for the water hammer excited response of piping structures</td>
<td>1.1</td>
<td>31.12.2017</td>
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<tr>
<td>D7.1.1</td>
<td>Project report on weld repair mock-up residual stress measurements or scientific publication on the subject</td>
<td>4.5</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D7.2.1</td>
<td>Project report on FEM studies of the plasticity produced by thermal cycles Removed due to budget cuts</td>
<td>0.0</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>D8.2</td>
<td>VTT Research report on practical utilization of research results.</td>
<td>0.4</td>
<td>30.6.2017</td>
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<tr>
<td><strong>Total pm</strong></td>
<td><strong>19.2</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
4. Project organisation

Juha Kuutti (VTT) will be the project manager. Deputy project manager will be Otso Cronvall (VTT). VTT will also be the responsible project organization. Project partner Aalto University will be responsible for WP7. TVO and the Swedish-Finnish Beräkningsgrupp (BG) are important industry partners and sources of external financing in the project but no specific deliverables are expected from TVO or BG. The researchers participating in the project are listed in the table below. The person month total exceeds the sum of person months for each deliverable as the project management and work group participations require resources but does not produce specific deliverables.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Juha Kuutti</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>WP1, WP4, WP8</td>
<td>2.8</td>
</tr>
<tr>
<td>Otso Cronvall</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>WP2, WP5</td>
<td>2.5</td>
</tr>
<tr>
<td>Tommi Seppänen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>WP3</td>
<td>2.9</td>
</tr>
<tr>
<td>Esko Arilahti</td>
<td>Senior Research Engineer</td>
<td>VTT</td>
<td>WP3</td>
<td>1.8</td>
</tr>
<tr>
<td>Jouni Alhainen</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>WP3</td>
<td>1.0</td>
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<tr>
<td>Jussi Solin</td>
<td>Principal scientist</td>
<td>VTT</td>
<td>WP3</td>
<td>0.3</td>
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<tr>
<td>Antti Timperi</td>
<td>Senior scientist</td>
<td>VTT</td>
<td>WP4, WP6</td>
<td>1.9</td>
</tr>
<tr>
<td>Ahti Oinonen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>WP5, WP6</td>
<td>2.4</td>
</tr>
<tr>
<td>Qais Saifi</td>
<td>Research scientist</td>
<td>VTT</td>
<td>WP2, WP5</td>
<td>0.8</td>
</tr>
<tr>
<td>Tero Tyrväinen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>WP5</td>
<td>0.7</td>
</tr>
<tr>
<td>Iikka Virkkunen</td>
<td>Professor</td>
<td>Aalto Uni.</td>
<td>WP7</td>
<td>1.0</td>
</tr>
<tr>
<td>Joonas Kähkönen</td>
<td>Researcher</td>
<td>Aalto Uni.</td>
<td>WP7</td>
<td>4.5</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>22.6</strong></td>
</tr>
</tbody>
</table>
5. Risk management

Most work packages in the project concern non-experimental studies. The typical risks related to this kind of studies are related to personnel resources. A large amount of resources are demanded for several key persons in the project group. The main risk of the project is that a key researcher in the project would be unavailable to carry out the planned research and a suitable backup is not readily found. In this case the replacing scientist may be unable to perform all planned work and some studies need to be postponed to next project years. For the experimental work in WP3, the risks are experimental failures and unexpected results. Also, an agreement over the supply of a relevant piping material for the tests has been reached but the material itself has not been delivered to VTT at the time of preparing this application. The risk of not receiving the material due to unexpected circumstances has to be recognized. Thorough understanding and interpretation of the results may require additional experiments or more resources than planned, which may not fit into the project budget. In this case the findings will be reported but not all hypotheses can be verified within this project.

The pipe sections to be studied in T7.4 are in service in OL1 and OL2 and were removed in revisions 2015 and 2016. By research of these pipe sections highly significant results are expected. Research of the pipe sections requires successful decontamination, even though the measurements are performed inside the Olkiluoto NPP facilities. Schedule and progress of the decontamination affects directly this project. This risk was realized in 2015 as the pipe sections were not available for measurements.

Contamination of the measurement equipment has to be avoided in order to get clearance after measurements. The hole drilling measurements are made with a drill, which releases a very small amount of debris from the studied specimens. The use of the drill is a necessity in order to get reliable residual stress data from the pipes. The cost of the drill is about 10 – 20 k€ in case of irreversible contamination. The X-ray measurement equipment does not release any material from the examined specimens, so the only risk (highly unlikely) is dust from the atmosphere.

The exact availability of the repair weld samples for residual stress measurements is currently unclear and there is a risk that some of the measurements cannot be completed as planned.
References


2.3 Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessel internals. IAEA-TECDOC-1471, International Atomic Energy Agency (IAEA), Vienna, Austria, October 2005. 107 p.

3.1 STUK, 2002. YVL-guide 3.5, Ensuring the strength of nuclear power plant pressure devices, issue 5.4.2002. (in Finnish, but translations exist)


5.2 http://safelife.jrc.ec.europa.eu/eniq/

5.3 ISI of nuclear facility pressure equipment with non-destructive testing methods. Guide YVL E.5, Radiation and Nuclear Safety Authority (STUK), 20 May 2014. 50 p.


6.3 ASME Boiler and Pressure Vessel Code, Section III - Construction of Nuclear Facility Components, American Society of Mechanical Engineers, USA, 2005.


7.1 Anon. Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessel internals. 2005. IAEA-TECDOC-1471. IAEA, Vienna. ISBN 92–0–109205–9


## Resource Plan for 2017

### Analysis of Fatigue and Other Cumulative ageing to extend lifetime

#### TOTAL BUDGET

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Amount</td>
<td>VYR (%)</td>
</tr>
<tr>
<td><strong>Volume</strong></td>
<td>person mo.</td>
<td>euro</td>
</tr>
<tr>
<td><strong>Personnel</strong></td>
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<td><strong>Travel</strong></td>
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<tr>
<td><strong>Ext serv</strong></td>
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<tr>
<td><strong>Other</strong></td>
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<tr>
<td><strong>TOTAL</strong></td>
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<tr>
<td><strong>VYR</strong></td>
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<td><strong>TVO</strong></td>
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<td><strong>BG</strong></td>
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<td><strong>VTT</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Other</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

| WP1 - Remaining lifetime and LTO of components with defects | 0.7 | 10 | 0 | 0 | 0 | 0 | 1 | 11 | 8 | 0 | 0 | 3 | 0 |
| T1.1 Evaluation of allowable crack sizes | 0.7 | 10 | 1 | 11 | 8 | 3 |
| T1.2 ETSON EG2 activities | 0.0 | 0 | 0 | 0 | 0 | 0 |

| WP2 - Susceptibility of BWR RPV internals to degradation mechanisms | 2.7 | 38 | 0 | 0 | 0 | 0 | 2 | 40 | 3 | 0 | 18 | 15 | 4 | 0 |
| T2.1 Susceptibility of BWR RPV internals to degradation mechanisms | 2.7 | 38 | 2 | 40 | 3 | 18 | 15 | 4 |

| WP3 - Fatigue usage of primary circuit | 4.6 | 65 | 11 | 3 | 0 | 0 | 17 | 96 | 65 | 0 | 0 | 0 | 31 | 0 |
| T3.1 Experimental Fatigue R&D in LWR Coolant Water | 3.4 | 47 | 7 | 3 | 13 | 70 | 48 | 22 |
| T3.2 Mechanism based model to justify revised Fitt | 1.3 | 18 | 4 | 3 | 28 | 17 | 9 |

| WP4 - Fatigue and crack growth caused by thermal loads | 2.1 | 30 | 0 | 2 | 0 | 0 | 4 | 36 | 22 | 0 | 0 | 0 | 14 | 0 |
| T4.1 Coupled CFD-FEM modelling of thermal fatigue | 1.1 | 16 | 2 | 18 | 11 | 7 |
| T4.2 Evaluation of low-cycle thermal fatigue cracking | 1.0 | 14 | 2 | 2 | 18 | 11 | 7 |

| WP5 - RI-III development and probabilistic safety assessment | 1.8 | 25 | 0 | 6 | 0 | 0 | 2 | 33 | 17 | 0 | 6 | 0 | 10 | 0 |
| T5.1 Probabilistic LBB assessment | 0.0 | 0 | 0 | 0 | 0 | 0 | 0 |
| T5.2 Extension and update of quantitative VTT RI-III analysis procedure | 0.9 | 12 | 1 | 13 | 4 | 6 | 3 |
| T5.3 Comparison between PRA and RI-III analyses | 0.9 | 9 | 1 | 10 | 6 | 3 |
| T5.4 NUGENIA TA8 (ENIQ TGR) activities | 0.3 | 4 | 6 | 0 | 10 | 7 | 3 |

| WP6 - Dynamic loading of NPP piping systems | 2.1 | 30 | 0 | 0 | 0 | 0 | 2 | 32 | 12 | 0 | 9 | 0 | 11 | 0 |
| T6.1 Review of water hammer analysis methods | 2.1 | 3 | 0 | 1 | 1 | 4 | 1 | 4 |
| T6.2 Combination of moments from dynamic analyses | 2.6 | 9 | 1 | 10 | 5 | 4 | 3 |
| T6.3 Development of water hammer response analysis methods | 1.0 | 14 | 1 | 15 | 9 | 6 |

| WP7 - Residual stresses in BWR NPPs | 5.5 | 58 | 3 | 2 | 0 | 1 | 0 | 64 | 49 | 19 | 0 | 0 | 0 | 0 |
| T7.1 Residual stresses due to repair welding | 5.5 | 58 | 3 | 2 | 1 | 64 | 49 | 19 |
| T7.2 FEM studies of the plasticity produced by thermal cycles | 0.0 | 0 | 0 | 0 | 0 |

| WP8 - Project management and review | 0.8 | 11 | 0 | 0 | 0 | 0 | 2 | 13 | 6 | 0 | 0 | 0 | 5 | 0 |
| WPM 1 Management | 0.4 | 6 | 1 | 7 | 4 | 1 | 3 |
| WPM 2 Utilization of results | 0.4 | 5 | 1 | 6 | 4 | 2 |

| **TOTAL** | 20.4 | 267 | 14 | 13 | 0 | 1 | 30 | 325 | 180 | 19 | 33 | 15 | 78 | 0 |
| **Comments:**
Person months for WPS 1-6 are an estimation calculated using 14k€/mo. Actual costs differ depending on individual salaries.
Research environment and program costs listed in “other” for WPs 1-6 and 8 (12 €/h for WP3, 5 €/h for others).
Planned travel costs are for ASME PVP 2017 (WP3.1) and SMiRT24 (WP4.2) conferences
Travel costs are allocated also for participation in NUGENIA TA8 in WP5.4
WPs 2.5,6,7 include both funding and in-kind contribution from TVO. Estimated total amount of in-kind work is 2.0 months.
## Analysis of Fatigue and Other Cumulative ageing to extend lifetime

### BUDGET FOR VTT WORK PACKAGES

<table>
<thead>
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<th>Work packages and Tasks</th>
<th>Volume</th>
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### Comments:
- Person months for WPS 1-6 are an estimation calculated using 14k€/mo. Actual costs differ depending on individual salaries.
- Research environment and program costs listed in “other” for WPs 1-6 and 8 (11 €/h for WP3, 3 €/h for others).
- Planned travel costs are for ASME PVP 2017 (WP3.1) and SMiRT24 (WP4.2) conferences.
- Travel costs are allocated also for participation in NUGENIA TAI and ETSON EG2 activities (WP5.4, WP1.2).
- WPs 2, 5, 6, 7 include both funding and in-kind contribution from TVO. Estimated total amount of in-kind work is 2.0 months.
### Analysis of Fatigue and Other Cumulative Ageing to Extend Lifetime

#### BUDGET FOR AALTO WORK PACKAGES

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**Comments:**
- Budget updated based on funding proposals 20.12.2016
- Aalto Uni. funding reduced proportionally to VYR funding
SAFIR2018 Project plan

LOST

Long term operation aspects of structural integrity

Sebastian Lindqvist
Project manager

Kim Wallin
Deputy project manager

Heikki Keinänen, Pekka Nevasmaa, Päivi Karjalainen-Roikonen, Juha Kuutti
1. Research theme and motivation

The updates for 2017 are described in section 2, work plan. The realized budget for 2017 was smaller than the requested budget, therefore, some of the deliverables described in the original plan can not be done or are reduced. The changes in the plan are described in section 2 under each task. Section 1, research theme and motivation, gives background information of the research topics in LOST.

The goal of SAFIR2018 subproject long term operation aspects of structural integrity (LOST) is to develop through experimental and numerical methods more accurate structural safety assessment methods to the nuclear power plant (NPP) end users and STUK.

1.1 Background and state-of-the-art

A systematic ageing management procedure is the basis for justifying the safe long term operation (LTO) of nuclear power plants. One fundamental part in this process is to assess the structural integrity of the NPP components such as reactor pressure vessel (RPV), pipes, welds and valves. In this project a comprehensive investigation is done considering the possibility of fast fracture in the upper shelf temperature regime of RPVs. Improved methods are developed for fracture toughness and crack driving force estimation of dissimilar metal welds (DMW). Also methods are developed for estimation of residual stresses in repaired DMWs, and the current methods for surveillance material testing are improved.

1.1.1 Advanced structural integrity (WP1)

Lifetime assessment of individual components and piping in nuclear power plants (NPP) is a mandatory part of every Periodic Safety Report as well as it is necessary for component/plant life management and potential plant life extension. In the same time, such assessment is also necessary for safe operation of components in NPPs.

VERLIFE – “Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs during Operation” was developed within the 5th Framework Programme of the European Union in 2003 and later upgraded within the 6th Framework Programme „COVERS – Safety of WWER NPPs“ of the European Union in 2008.

Until the VERLIFE preparation, no legal procedures or standard guidelines existed for lifetime/integrity assessment of components and piping in operating NPPs of WWER type. Former Soviet rules and standards had been prepared and approved only for design and manufacturing stage of NPPs. These rules/standards mostly are not applicable for operating plants or they need some modifications and extensions to be usable also for operating components. Approaches used in WWER Codes and standards are in some parts different than those applied in Western PWR ones, thus a comparison of lifetime assessment using these two types of codes could be different and non-comparable.

The main goal of the project VERLIFE has been in a preparation, evaluation and mutual agreement of a “Unified procedure for Lifetime Assessment of Components and Piping in VVER Type Nuclear Power Plants”.

The procedure is largely based on former Soviet rules and codes, as WWER components were designed and manufactured in accordance with requirements of these codes and from prescribed materials.

In preparation of the “Unified Procedure” the following principles and inputs were agreed: VVER components were designed and manufactured in accordance with former Soviet rules and standards, IAEA activities in the field of VVER components integrity assessment, Approaches applied in PWR components integrity and lifetime evaluation, last developments in fracture mechanics and their application to component integrity.

International organizations like ASME does not exist in VVER countries (only similar national organizations exist in some countries), and procedure for Russian Codes preparation is quite different – they are prepared by special organisations under the contract from Regulatory Bodies and/or utilities.

VERLIFE can be used for evaluation of residual lifetime of components and piping of NPPs with VVER type reactors designed, manufactured, inspected and put into operation in accordance with former Soviet Rules and Codes. It can be used for an elaboration of Periodic Safety Reports (or similar type of documentation) to demonstrate operational safety and reliability of components and piping during reactor operation.

VERLIFE is periodically upgraded and revised with a 3-4 years period. It is extended for the application to other type of components and integrity evaluation and harmonization of procedures for lifetime assessment of components and piping in VVER NPPs during operation. A new revision round of VERLIFE is presently beginning and it may have an impact on the Lovisa I & II and Hanhikivi I reactors. The aim of the new revision round is to validate fracture toughness based trend line curves for VVER RPVs.

Another item imperative for the long term operation of RPVs is the use of miniature fracture toughness specimens, since the standard specimens used in surveillance programs have already been spent. VTT has already participated in one international round robin dealing with miniature size C(T) specimens, but the round robin activities continue with the aim of standardisation and evaluation guides of miniature size C(T) specimens.

Connected to this topic, the Swedish Nuclear Utilities Materials Group, MG, has evaluated the possibilities to utilize the shutdown Barsebäck nuclear reactors as a platform for further studies of the effect of ageing on materials and components in a project named BREDA. The Barsebäck core weld has a high Ni content, making it also basically representative of VVER-1000 materials. Destructive sampling is proposed to acquire irradiated material from the RPV wall of Barsebäck to enable studies of the properties in areas subjected to (relatively speaking) high and low dose levels. This is complemented with surveillance material covering doses up to $9 \times 10^{19}$ corresponding to a Charpy-V shift in the range of 150°C. The broken surveillance specimens can be tested either using reconstitution techniques or by using miniature C(T) specimens (or both). Experiments carried out in BREDA decrease the uncertainty between experimental results from Charpy-V and fracture toughness specimens, and between predictions made by using the surveillance material and the real aging behaviour of the RPV.

The work done in BREDA is carried out between Royal Institute of Technology-KTH, Stockholm, Chalmers Technical University – CTH, Göteborg and VTT Technical Research Center of Finland, Espoo. VTT is responsible for mechanical testing (tensile, Impact and fracture mechanical testing). A big part of the costs of mechanical testing at VTT is currently being included as support to a pilot project proposed under the auspices of Energiforsk (Swedish energy research). However, negotiations of the costs are on-going and in 2016 there is still some work to be done considering the feasibility of the project.
Connected to the ageing of the material, the YVL guideline E.4 important for RPV safety, entitled “Strength analyses of nuclear power plant pressure equipment” contains in chapter 6 “Brittle fracture analysis”, section 6.8 “Other fast fracture considerations” the following paragraph:

616. 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. This could occur in thick-walled components which undergo rapid cooling under high pressure. The adequacy of the toughness values of the upper shelf shall be analysed, where necessary. The methods and criteria used are subject to STUK’s approval.

The text in paragraph 616 is quite short and the term “fast fracture” in the context of “upper shelf area” may be open for different interpretations. Fast fracture (catastrophic failure) in the upper shelf area occurs either by tearing instability (or plastic collapse) or brittle fracture. In the latter case the material has not been in the actual upper shelf temperature zone, due to rapid cooling that can be caused by a LOCA event. Fast fracture in the upper shelf area is also directly related to leak before break (LBB) assessment.

The probability of fast fracture in the upper shelf temperature regime is affected, in addition to the crack driving force, by changes in constraint, the loading rate and the increased sampling due to crack propagation. Crack propagation, per se can lead to constraint changes and it also affects the effective strain rate at the crack tip. Especially in the case of LBB where a surface crack transforms instantaneously to a through-wall crack the increase in local strain rate, combined with constraint change and ductile crack extension, can be sufficient to cause a transition from ductile fracture to brittle fracture. This and other similar events require the development and verification of an advanced Master Curve procedure to account for combined, constraint, ductile tearing and loading rate effects on the brittle fracture probability.

Physically, upper shelf is defined as the temperature range where brittle fracture cannot occur. Technologically, however, the definition of upper shelf or onset of upper shelf is much more controversial. The upper shelf definition that is probably closest to the physical definition is the plastic fracture transition temperature (FTP) determined with the explosion bulge test, but even then it is related to the tested section size (normally less than 25 mm). Thus it does not represent a definitive upper shelf temperature. Historically, one simple definition of upper shelf was the temperature where a Charpy-V impact test leads to a 100 % ductile fracture surface. The old British requirement that the British nuclear reactors have to operate on the upper shelf stems from this definition. The introduction of fracture mechanics and fracture toughness tests has, however, clouded the technological definition of upper shelf.

In the USA it is common to define the onset of upper shelf as the temperature where the fracture toughness exceeds the ductile initiation fracture toughness \(J_{IC}\). Similarly, in Britain the onset of upper shelf temperature (OUST) is presently defined as the intersection between the 5% probability curve for fracture in the transition region and the mean curve for ductile fracture initiation at 0.2 mm crack growth. The USA definition does not specify a specific probability for the brittle fracture estimate and the ductile initiation definitions differ slightly. The American \(J_{IC}\) corresponds to 0.2 mm ductile tearing in addition to crack tip blunting, whereas the British definition is 0.2 mm total crack tip extension including blunting. For a high toughness material, the British definition may correspond purely to crack tip blunting.

These definitions, however, rule in no way out the possibility of brittle fracture after some ductile tearing (fast fracture) at a higher temperature. Also, the OUST temperature depends, besides on the brittle fracture properties of the material, also on the ductile fracture properties. For the same brittle fracture properties, a material with a
poor ductile fracture resistance will show a lower OUST than a material with a high ductile fracture resistance. The OUST is also affected by loading rate and constraint.

Because of the complexity of the catastrophic fast fracture events in the transition region, a comprehensive study is needed. Two specific open research topics are identified. 1) Ductile crack growth during a temperature transient and 2) A constraint, loading rate and crack growth adjusted modified advanced Master Curve methodology.

1.1.2 Dissimilar metal welds, DMW (WP2)

Dissimilar metal welds (DMW) are critical components of NPPs, because numerous flaws have been detected in these areas with NDE and DMWs contain regions that are prone to fracture. The latest NDE methods tend to give more detailed information of existing defects/flaws in components. Flawed components can be repaired. When long term operation is considered of repaired components, an essential point is how to assess the usability and remaining lifetime that are affected by the residual stress state in the component.

The overlay welding of components, e.g. bimetallic welds in nozzles, is a well-established repair technique in USA, especially in the case of circumferential flaws/cracks. The recent results [1,2] show, however, that the situation may be different in the case of deep axial defects, for which overlay welding may lead to an unfavourable residual stress state. Most of the present computational approaches to model these are based on axisymmetric models. Three dimensional modelling may give more accurate results as compared to those obtained with axisymmetric models [2]. Therefore, methods based on three dimensional modelling should be developed for assessment of residual stresses in repaired DMWs with axial defects.

Another issue for DMWs is the development of descriptive fracture toughness and crack driving force determination techniques. Accurate fracture toughness determination of the most critical region in DMWs is important for LBB analysis that is used to ensure structural integrity. In addition, to further increase the accuracy of the LBB assessment of DMWs advanced numerical equations are needed to precisely determine the crack driving force in DMWs. The development of accurate fracture toughness and crack driving force determination techniques require both experimental and numerical work.

Descriptive fracture toughness values for DMWs are not easily achieved, because the current standards to measure the fracture toughness are only developed for homogeneous materials. In DMWs the deformation of the material in front of the crack can be concentrated in a completely different manner than in homogeneous specimens. This difference in deformation behaviour can make the homogeneous equations inapplicable for heterogeneous welds. Therefore the current standards can not necessarily be directly applied to heterogeneous materials. Another problem is that in DMWs cracks can deviate away from their initial fracture plane. There is no standard that tells how to take into account crack deviation in fracture toughness measurements.

The problems in crack driving force (CDF) determination for DMW components lie in the heterogeneity of the weld. A specific challenge of DMWs in this respect is the asymmetry of local stress and, consequently strain distributions, owing to the existence of base metals and narrow microstructural regions with significantly different mechanical properties and hence stress-strain characteristics [3]. The current CDF J-integral solution is developed for homogeneous materials and its applicability to heterogeneous structures is not known. The CDF calculated for DMWs can not thus be considered accurate.

To overcome these difficulties in fracture toughness measurements and crack driving force determination of DMWs both numerical and experimental investigations are required. Experimental methods are required to understand the material behaviour, determine accurate FE models and calibration parameters for the numerical
work. Numerical work is required to improve the analytical fracture toughness solutions and to improve the formulas for crack driving force in DMWs. This work package utilizes data and material from project MULTIMETAL, a European collaboration project related to DMWs.

1.2 Objectives and expected results

In WP1, advanced structural integrity, the objective is to develop new advanced structural integrity methods to describe the ductile crack growth during a temperature transient accounting for temperature history effects and for fast fracture. The second objective is to develop a constraint, loading rate and crack growth adjusted modified advanced Master Curve methodology to deal with complex events related e.g. to leak before break (LBB) assessment. These advanced evaluation methods influence also the accuracy of RPVs structural integrity. This topic increases the knowledge of fast fracture in the upper shelf temperature, which is also required in the YVL-guidelines.

Presently, the reactor pressure vessel embrittlement trend curves are based on the Charpy-V test, even though it is well known that it does not describe reliably the true fracture toughness shift. The primary objective in the planned work connected to VERLIFE is to develop fracture toughness based trend curves for WWER materials. This has a vital importance for the long term operation of nuclear reactor where the reliable knowledge of the available margin between crack driving force and fracture toughness may become critical. It also enables an assessment of the expected embrittlement trend of reactors in the planning stage.

Presently, the most promising miniature specimen for brittle fracture toughness testing is the 4 mm C(T) specimen proposed by CRIEPI. The objective is to provide additional required validation of the specimen and to modify the specimen to be optimally more suitable to use with broken Charpy-V specimen. The expected result is a new validated miniature specimen requiring a minimum amount of machining.

The goal with the BREDA project utilising the Barsebäck surveillance material and possible trepan from the pressure vessel is to examine more closely the embrittlement measured with Charpy-V versus the actual fracture toughness and to examine the representativity of surveillance material with respect to the actual pressure vessel. This representativity has recently been recognized as a major source of uncertainty in the structural integrity assessment. The experimental programme done in BREDA is expected to increase the accuracy of RPV ageing predictions based on fracture toughness data.

The objective in WP2, residual stresses, is an enhanced treatment of weld residual stresses in repaired DMWs, and utilisation of residual stresses in fracture assessment. Improvements in residual stress estimation increase the accuracy of the estimation of the remaining life.

Secondly, experimental and numerical investigations are used in WP2 to develop the calculation methods for crack driving force and fracture toughness in DMWs. The aim of the experimental investigations is to retrieve a deeper understanding of the fracture behaviour in DMWs interface regions, develop a method to characterise the near-interface zones (NIZ) of DMWs and to produce data for the numerical investigations. The results of the experimental characterisation are used for calibration of FE models that are used to improve the analytical solutions of fracture toughness and to develop crack driving force solutions for NPP DMWs. These objectives impact the methods used to estimate the critical crack size. The results can be incorporated into the LBB analysis and leakage probability calculations in future projects. Results and mock-up from European collaboration project MULTIMETAL are exploited in WP2.
1.3 Exploitation of the result

The results can be used for structural integrity assessment and fracture mechanical analysis of the reactor circuit in NPPs by safety authorities and nuclear power plant end users. Improved structural integrity assessment can also be applied in design of new power plants to ensure the necessary safety. The results improve and clarify the YVL instructions and are applicable immediately after the completion of the different tasks.

The tasks in WP1 are related to the YVL-guidelines. The end users of the results will be STUK, plant operators and manufacturers in need of methods to show the plant structural safety as prescribed in the YVL guides related to fast fracture in RPVs. Fast fracture can occur in thick-walled components which undergo rapid cooling under high pressure e.g. LOCA. The results increase also the applicability of LBB in NPPs.

The miniature fracture toughness specimen, VERLIFE and BREDA tasks in WP1 improve the methods used for assessing structural safety of RPVs. Validating the use of miniature fracture toughness specimens decreases the material consumption in surveillance testing, thus enabling assessment of RPV behaviour of a longer period. In BREDA project the current fracture toughness methods for assessing the ageing of RPV is verified against Barsebäck RPV. The goal of VERLIFE is to develop a Master Curve based irradiation embrittlement trend curve for VVER materials. The Master Curve can be exploited in assessments of the VVER RPVs ageing instead of quantitative methods based on charpy-V.

The results achieved in WP2 are useful for assessing the possibility of using thick overlay welding as a repair technique for axial flaws in DMWs. In addition methods for assessing residual stresses caused by internal machining and subsequent welding of the machined region/volume in DMWs are developed. The results achieved in WP2 will increase the accuracy of residual stress assessments in repaired DMWs. This has an impact on the life time estimations e.g. inspection interval estimation for cracks in NPP structures. The increased accuracy in life time estimations is useful for STUK and the plant operator.

The improvements done considering the fracture toughness and crack driving force estimation of DMWs have a direct effect on accuracy of LBB analysis and on leakage probability calculations. The end users benefit from these results by getting more precise methods for estimation of the structural performance of pipes.

1.4 Appropriateness of the project to SAFIR2018 programme

One of the main goals of SAFIR2018 is to improve life time management of NPPs. In LOST this goal is fulfilled by developing the current safety assessment methods further to ensure the reliability of structural materials. The results of LOST affect LBB assessment, by increasing the current understanding of fast fracture and introducing more accurate methods to calculate J-integral and fracture toughness in DMWs, and by taking into consideration residual stresses in welds. Also methods are developed for assessing structural safety of RPVs in form of improved surveillance material testing techniques and analytical solutions. These developments affect the life time management of NPPs in the design and operation stages, and increase the reliability of long-term use of NPPs.

Other important goals in SAFIR2018 are: knowledge transfer of safe and economical operation of nuclear power plants to young scientists, development of the state-of-the–art and participation in international cooperation. These three points are fulfilled in LOST.

Knowledge transfer is done by involving and creating opportunities for young research scientists and engineers. Three young research scientists work in this project (Sebastian Lindqvist, Qais Saifi, Laura Sirkiä). The training of young scientists is described in more detail in section 1.5.
Developing the current state-of-the-art is done by using the experience from previous projects to create new research topics. The main goals of the research are described in section 1.2.

The international cooperation is diverse. WP1, advanced structural integrity, is incorporated into an international project called ATLAS. Results and experience of MULTIMETAL is used in WP2. NUGENIA cooperation is continuing in this project. Cooperation will be done with Bay Zoltan foundation, a Hungarian institute, in the area of VVER DMWs. A concrete outcome of the project is articles that are written as a part of the doctoral theses and international conferences and seminars to present the results.

SAFIR2018 framework program plan defines three main research areas (plant safety and system engineering, reactor safety, structural safety and materials). These main research areas are further categorized into subareas containing different problems. LOST is mostly related to the main research area about structural safety and materials. This main research area is further divided into seven subareas. Of these areas the most relevant topics for LOST are assessment of failure and damage mechanisms, advanced assessment methods for structural safety, new material solutions, and life cycle management methods and life cycle extension. These topics are marked in figure 1 with green colour.

![Figure 1. Appropriateness of LOST to SAFIR2018 programme](image)

The topic about assessment of failure and damage mechanisms addresses the need to develop the knowledge of fracture mechanics and its modelling in nuclear power plants to secure durability and structural integrity of the older and new plants with more accurate and diverse computational analyses. Especially, reactor pressure vessel steels and stainless steel pipeline materials including their welds and dissimilar metal welds are important. In WP1, advanced structural integrity, and WP2, dissimilar metal welds, the previously mentioned challenges are
approached by conducting an excessive experimental program on fracture mechanical characterization of fast fracture and dissimilar metal welds to develop the current understanding and methods for failure analysis. In WP2, dissimilar metal welds, also numerical modelling of fracture in DMWs is done by local approach method and the results are verified before use. The transferability of the results retrieved from modelling and experimental analysis to real structures is considered.

The subarea composing of advanced assessment methods for structural safety emphasises the importance of developing LBB methods to achieve adequate reliability for pipes in, especially, new NPPs. In addition, more realistic evaluation approaches of loads caused by different operational conditions are needed. In WP1, advanced structural integrity, results of fast fracture investigations are directly connected to LBB. In addition, fracture mechanical characterization in WP2, dissimilar metal welds, provides basis for further assessment of DMW subcritical crack growth behaviour, which is important for LBB. In WP2 methods for realistic estimation of residual stresses in NPPs and crack growth under residual stresses will be developed.

New material solutions are described in SAFIR2018 as a topic that includes manufacturing techniques and structural solutions used in the new nuclear power plants in Finland. In WP 2, dissimilar metal welds, advanced material characterization methods are used for NPP DMWs. The research topics in the project can incorporate materials that become relevant during the project.

1.5 Education of experts

In this project experts are trained to the NPP safety area by involving young research scientist into different tasks and deepening the knowledge of senior scientists. Doctoral theses are done by young research scientists Sebastian Lindqvist and Qais Saifi, and one by senior scientists Päivi Karjalainen-Roikonen. Laura Sirkiä is doing her Master’s thesis to this project. The topics of the PhD theses are connected to DMWs and temperature transition effect on ductile crack growth.

Secondly, the project manager is a young research scientist. The experience and knowledge of older scientists in project managing is important to pass down to young scientists. Thirdly, a generation change is in progress in experimental fracture mechanical testing. To transfer the experience in execution of complicated and demanding testing procedures is crucial for accurate testing. Fourth, the international cooperation in the project provides a forum for young research scientists to create international connections, bring expertise from abroad to Finland and learn to know important international operators and forums.

1.5.1 International cooperation

Taking part in international networks, meetings and conferences in the field of fracture mechanics and structural integrity is an essential for education of experts and providing possibilities for younger scientist to grow and mature into the international co-operation. Previous cooperation within international networks in the area of structural integrity and fracture mechanics has enabled participation in standardization bodies, such as ASTM. Contribution has been very fruitful resulting in international Master Curve standard ASTM E1921. It is of crucial importance that this work will continue within this project. This applies to other networks as well.

Participation in the ASTM (American Society for Testing and Materials) E08 Fatigue and Fracture executive committee as well as in the international networks such as IGRDM (International Group for Radiation Damage Mechanisms) and IAEA (International Atomic Energy Agency) are realised within the project. In European perspective, participation in NUGENIA network is emphasised. Research results will be published at relevant papers and at major international conferences such as ASME PVP (Pressure vessel & piping), SMRT (Structural Me-
The project is directly connected to EU Horizon2020 project ATLAS (Advanced Structural Integrity Assessment Tools for Safe Long Term Operation). The ATLAS will be coordinated by VTT and the consortium consists of 21 organisations from 8 EU member states and two collaborating countries (USA and Japan) each with a high percentage of nuclear power in the total national electricity production. Specifically, ATLAS project will focus on:

- innovative quantitative methodologies to transfer laboratory material properties to assess the structural integrity of large components,
- an enhanced treatment of weld residual stresses when subjected to long term operation,
- advanced simulation tools based on fracture mechanics methods using physically based mechanistic models,
- improved engineering methods to assess components under long term operation taking into account specific operational demands,
- integrated probabilistic assessment methods to reveal uncertainties and justify safety margins.

ATLAS will have a significant impact on the safety of operational nuclear power plants. The project will demonstrate and reveal inherent safety margins introduced by the conservative approaches used during design and dictated by codes and standards employed through-out the life of the plant. The outcomes from ATLAS will therefore support the long term operation of nuclear power plants. This will be achieved by using more advanced and realistic scientific methods to assess the integrity piping. ATLAS will provide evidence by to support the methods by carrying large scale tests using original piping materials. Researcher exchange is planned specifically within the ATLAS project.

Training of new experts inevitably requires work in the international research environment. Within this project there will be many senior experts in structural integrity who are brought together with young scientists. They have the opportunity to collaborate on many common technical subjects. Therefore, there are unique training possibilities such as materials testing, computational modelling and applied engineering. By interacting with the experts young scientists can be educated and become familiar with these topics. To enhance this, exchange activities will be carried out in the international projects such as ATLAS to send young scientists to host organisations for training. The activities will be enhanced through links with the NUGENIA network.
2. Work plan

An overall plan for 2015-2018 and a specific plan for 2017 are described next. The 2017 plans are described in the subsections after the overall plan for 2015-2018.

2.1 Advanced structural integrity (WP1)

To ensure long term reliable use of reactor pressure vessel, the current structural integrity methods are improved in work package 1 of LOST. The goal is to develop our current understanding of I) fast fracture in the upper shelf region, task 1.1, II) appropriateness of surveillance programmes, task 1.2, and III) miniature fracture toughness specimens, task 1.3.

Task 1.1 focuses on fast fracture in upper shelf region. Similarly as brittle fracture, fast fracture in upper shelf region can have catastrophic consequences. Fast fracture in upper shelf region has not previously been investigated, even if YVL guideline states that “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”.

Task 1.1, crack growth in temperature transients, is divided into 4 tasks:

- Task 1: Determine the temperature dependence of ductile fracture tearing resistance in the temperature range 20°C...300°C. (2016)
- Task 2: Determine the effect of temperature history on tearing resistance. (2016)
- Task 3: Determine the effect of in situ increase or decrease of temperature on tearing resistance. (2017)
- Task 4: Develop methodology to account for the effect of temperature transients on ductile tearing resistance. (2018)

- In 2016, an experimental testing programme was executed and the results were analysed to complete task 1 and 2, above. 2017 activities are described in section 2.1.1.

In task 1.2, BREDA, the main goal is to estimate how well the surveillance programme describes the ageing behaviour of reactor pressure vessel. In BREDA, Barsebäck REsearch&Development Arena, samples are harvested from Barsebäck (Sweden) reactor pressure vessel. The fracture mechanical properties of the RPV samples are compared to the existing results from surveillance programme of Barsebäck. The experimental investigations focus on the weld, as the weld materials are the limiting materials from a Long term operation, LTO, perspective. VTT’s role in the project is to do the fracture mechanical tests and analyses (realised in LOST task 1.2) and to estimate the effect of microstructure on fracture toughness (realised in THELMA). The sample removal from the RPV is financed by Forsmark and Ringhals.

Currently:

- The Charpy-V surveillance samples from Barsebäck have been tested
- Two sample sets of relevant material are currently located at Areva, in Erlangen, Germany. The other set has been subjected to a dose that corresponds to 29 operation years of BWR (dose 1.03E+18, 1>MeV)
and the other set has been subjected to a dose that corresponds to 200 operation years of BWR (dose 5.90E+19, 1>MeV).
- Trepans from the Barsebäck reactor pressure vessel will become available by the end of 2017/beginning of 2018.

Timetable of the BREDA project:

- 2016: Mechanical testing and analysis of reference material
- 2017: Transportation of test material from Germany to VTT. Pre-investigations in SAFIR 2018, (see section 2.1.2). Microstructural characterisation of the reference material.
- 2018: Testing and analysis (mechanical and microstructural) of the material from Germany. The pressure vessel will not become available for trepan removal before the end of 2017 beginning of 2018 at earliest, after which the material transport can take place. The trepan removal is financed by Forsmark and Ringhals.
- 2019-2020: Testing (mechanical and microstructural) of the trepan and estimating the representability of the surveillance programme. Testing of trepan can be included to SAFIR2022.

- In 2016, miniature fracture toughness tests and analyses were done on the reference material of the Barsebäck RPV weld.

Miniature fracture toughness specimens are used in fracture toughness tests of the BREDA material, barsebäck weld, to further develop the miniature fracture toughness testing methods. The benefit with miniature fracture toughness specimens is that fracture toughness based data, more relevant from structural integrity perspective, can be obtained from already tested Charpy-V specimens. From one Charpy-V sized specimen one can get 8 miniature fracture toughness sized specimens. Another benefit is that the duration of surveillance programmes can be increased, due to reduced material consumption.

- In 2016, a conference paper was written to pressure vessel and piping conference about the applicability of miniature fracture toughness specimens.
- In 2016, tests on miniature fracture toughness specimens were done on reference weld metal from Barsebäck NPP. The goal is to investigate the applicability of miniature fracture toughness specimens for welds. The results are reported in Master’s thesis of Laura Sirkiä.

The aim in T1.3 (VERLIFE) is for VTT to participate in the VERLIFE project. In 2018-2019 the main aim is to develop a Master Curve based irradiation embrittlement trend curve for VVER materials.

2.1.1 Crack growth in temperature transients (T1.1)

The work done in task 1.1 is related to Päivi Karjalainen-Roikonen’s doctoral thesis.

In 2017, in deliverable 1.1.1, a VTT report is written based on fracture resistance testing in decreasing temperature gradients. The work includes fracture mechanical testing and analyses. Estimated duration of the work is 3,5 person months. The results are reported as a VTT report.

For a validation of the results, VTT participates in a national round robin related to brittle and ductile fracture toughness measurements. The main goal is to ensure the correctness of the fracture toughness measurements, aids also T2.2. This validation work is not realised in 2017, due to cuts in the budget.

2.1.2 BREDA (T1.2)

In 2017, the objectives for deliverable 1.2.1 are the following:
- Estimation on how representative the Russian steels, with similar Ni and Mn content but higher Cr content, are compared to Barsebäck weld.
- Verification of the $T_0$ and $T_{41J}$ dependence
- Estimate the effect of neutron spectrum on irradiation embrittlement in PWR and BWRs.

Estimated duration of the work is 3,5 person months. The results are reported in a VTT report.

In 2017, in deliverable 1.2.2, a fractography investigation is done to miniature C(T) specimens tested in 2016. The investigation focuses on the effect of initiation location in miniature C(T) specimens made of a reference weld material from Barsebäck NPP. The work is done in collaboration with THELMA project. The results are reported in Laura Sirkiä’s Master’s Thesis planned to be finished in November. Estimated duration of the work required for fractography analyses and reporting is 1 person month. Kim Wallin is also participating in two ASTM Committee E08 Fatigue and Fracture committee meetings in 2016, held in Toronto and Atlanta, focusing on the standardisation of miniature specimen testing needed for the Breda work.

The work done in deliverable 1.2.2 is mainly related to Laura Sirkiä’s Master’s thesis.

2.1.3 VERLIFE (T1.3)

VERLIFE project is still in the planning stage and at the moment a kick-off meeting is not necessary held in 2017. However, an option for reorganising the resources in LOST is kept open so that VTT can participate in the kick off meeting, if the VERLIFE collaboration starts in 2017. Estimated duration of this work is 0,25 person months and the outcome of the clarification is reported at SAFIR2018 reference group 5 meeting.

2.2 Dissimilar metal welds, DMW (WP2)

In work package 2, dissimilar metal welds, the aim is to develop methods and preparedness to analyse dissimilar metal welds, in case a crack is found from the weld. Dissimilar metal welds are prone to flaws, and if a flaw is found, then one has to understand the underlying physics. WP2 aims at increasing the physical understanding of (i) residual stresses, task 2.1 (ii) fracture, task 2.2 and (iii) crack growth, task 2.3, in DMW.

In task 2.1, procedures for calculation of residual stresses in repaired or replaced DMWs are developed. Estimation of these stresses is highly important for safe long term operation. The residual stresses are estimated with three dimensional numerical models. The work started in 2016 is continued in 2017 (see section 2.2.1).

- In 2016, a literature survey of the inlay welding was performed mainly in a mechanical point of view. The residual stresses and related structural phenomena related to inlay welding presented in literature were studied. The stresses due to thick overlay welding and combination of overlay and inlay welding were computed.

In task 2.2, the fracture mechanical assessment methods are improved for DMWs. Firstly, one goal is to develop DMW-specific equations for calculating fracture toughness. The current equations are developed for homogeneous metals. Second, the understanding of non-planar crack growth and its effect on fracture toughness is developed. Third, the sensitivity of specimen configuration on tearing resistance is investigated. The loading condition in front of the crack in standard specimen configurations can be overly conservative compared to real components.

- In 2015 a literature survey about $\eta$-factor in welds and DMWs was carried out.
- In 2015 a conference article was written. The conference was held in 2016 in Catania, European conference on fracture.
- In 2016 the effect of crack path on tearing resistance of DMWs was investigated
- In 2016 the effect of improved equations for DMWs was estimated. However, the improved equations were derived with an indirect method. In 2017 a better method is used to derive DMW specific equations (see Material characterisation (T2.2)).
- In 2016 two scientific articles were written.

In task 2.3, methods for crack driving force evaluation in DMWs are developed further from the currently existing ones, so that they can be reliably applied to DMWs. This enables one to obtain a descriptive estimate of crack driving force. Numerical methods used in this task will exploit experimental results from other tasks of LOST, T2.2 and T1.1, as well as utilising the extensive database and experience gained from previous international (BIMET, ADIMEW, STYLE, MULTIMETAL, PERDI) and national (SINI, FAR) projects.

- In 2016, crack growth computation for two CT specimens were carried out by GTN model. The specimens were made of dissimilar metals. 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.

2.2.1 Residual stresses (T2.1)

In 2017, in deliverable 2.1.1, the local repair together with inlay welding is studied. The existing data from literature has been collected in 2016 and current analysis methods are applied for modelling a local repair and inlay welding case performed for a full scale mock-up. The computational results are verified by the experimental measurements carried out in WP 6 of Found project. The results are reported as a VTT report and estimated duration of this work is 1,7 person months. Due to cuts in the budget for 2017, the possibility to utilise e.g. symmetry conditions and other modelling techniques are not estimated this year.

2.2.2 Material characterisation (T2.2)

The work done in T2.2 is related to Sebastian Lindqvist’s doctoral thesis.

The current ASTM E1820 equations used for calculating fracture toughness of dissimilar metal welds, a heterogeneous structure, are developed for homogeneous materials. A way to develop equations more suitable for heterogeneous structures is to derive DMW specific η-factors (eta-factors), a parameter in the fracture toughness equations.

In 2017, in D2.2.1, η-factors as a function of crack location and mismatch state are derived with finite element method. Three, different mismatch scenarios are considered; one based on the physically measured values, and in the other two cases the sensitivity of η-factor on the strength mismatch state is considered. The analyses focus on the interface region of a DMW. The effect of the new η factors are estimated by recalculating fracture toughness, and comparing new and old results. A scientific article is written of the results. Estimated duration of the work is 1,6 person months.

A non-standard fracture toughness specimen, single edge tension (SE(T)) specimen, can be more suitable for fracture toughness measurements of pipes than standard SE(B) and CT specimens, since the stress and strain condition in front of the crack is similar to the condition in components. In 2016, dissimilar metal weld SE(T) specimens were analysed in LOST. The results (J-R curves) suggested negative crack growth, which is physically impossible, and over 0,8 mm difference in measured and calculated initial crack size.

In 2017, in D2.2.2, the reason for these negative side effects of SE(T) specimens is investigated, by doing fracture mechanical tests on eight SE(T) specimens of a pressure vessel steel. The unloading compliance method is used. Each specimen is grown to a different crack length to understand the physical differences between measured and calculated results, and the negative crack growth. A scientific article is written of the results. Estimated
duration of the work is 1.7 person months. From the two articles, 0.6 person months were reduced due to cuts in the budget, which can have an effect on the completion of the work in 2017.

In 2017, in D2.2.3, Sebastian Lindqvist is participating in 14th International Conference on Fracture (ICF14) as the representative of Finland. The conference paper describes the use of an effective elastic modulus for cracks at the interface of a DMW, and how the modulus affects the calculated crack size. This conference article supports the activities planned for D2.2.2. Estimated duration of the work is 0.5 person months. The results are reported as a conference article.

2.2.3 Local approach (T2.3)

The work done in T2.3 is related to Qais Saifi’s doctoral thesis.

In 2017, in D2.3.1, the aim is to develop methods for predicting crack growth in components by deriving a new equation dependent on void size and strain state. Later on this equation is implemented into numerical models. The work is divided into three categories.

Experimental determination of void size. The goal is to:
- Experimental determination of void size as a function of strain
- Experimental measurement of fracture toughness (realised in task 1.1).

Analytical function development. The goal is to:
- Based on experimental results, a void growth function depending on strains will be developed.
- The void growth function will be injected into the GTN model in order to make more complete the model. This will become modified GTN model.

Numerical implementation. The goal is to:
- An Abaqus subroutine in FORTRAN will be written to insert modified GTN model as a constitutive equation for crack growth analyses.
- Numerical prediction of fracture toughness and crack growth with modified GTN model.
- Comparison between numerical and experimental results.

Estimated duration of the work is 4.1 person months. The results are reported in a VTT report.

D2.3.2, a scientific article written based on the results from D2.3.1, is not realised in 2017, due to cuts in the budget. Estimated duration of the work was 1 person month.

2.3 Fracture standard development and quality assurance (WP3)

The tasks related to this work package were removed from the 2017 plan, due to cuts in the budget.
## 3. Deliverables 2016

Table 1. Deliverables that are done in 2016.

<table>
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<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
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<td>D1.2.2</td>
<td>Master’s Thesis: Applicability of miniature fracture toughness specimens for welds</td>
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<td>VTT research report: Residual stresses in case of local repair together with inlay welding</td>
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<td>Scientific article: Improved equations for determination of fracture toughness of dissimilar metal welds</td>
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<td>D2.2.2</td>
<td>Scientific article: fracture toughness determination with SE(T) specimens of dissimilar metal welds</td>
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<td>D2.2.3</td>
<td>Conference article to 14th International Conference on Fracture: An effective elastic modulus in fracture toughness determination of cracks at the interface of a DMW</td>
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4. Project organisation

Project manager is Sebastian Lindqvist. VTT is responsible for the project.

<table>
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<tr>
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<th>Estimated person months (2017)</th>
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<tr>
<td>Sebastian Lindqvist</td>
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<td>VTT</td>
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<tr>
<td>Kim Wallin</td>
<td>Deputy project manager, Research professor</td>
<td>VTT</td>
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<td>Päivi Karjalainen-Roikonen</td>
<td>Principal scientist</td>
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<td>Qais Saifi</td>
<td>Research scientist</td>
<td>VTT</td>
<td>T2.3, T2.2</td>
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<tr>
<td>Juha-Matti Auto</td>
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<td>VTT</td>
<td>T1.2</td>
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<td>T2.1</td>
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<tr>
<td>Esa Varis</td>
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<td>VTT</td>
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<tr>
<td>Marke Mattila</td>
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<tr>
<td>Jorma Hietikko</td>
<td>Senior Research Technician</td>
<td>VTT</td>
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<td></td>
<td><strong>17.6</strong></td>
</tr>
</tbody>
</table>
5. Risk management

The project work might be delayed, due to delivery problems related to DMW mock-ups or other material. This risk is minimized by using reliable suppliers and part of the material used in LOST is already in possession of VTT. Additionally, loss of experienced staff can remarkably effect the execution of the project. The risk is minimized at VTT by programmes that include knowledge transfer from experienced scientist to younger scientist and by creating other opportunities for young scientists to develop their knowledge.
References


## LOST

### Long term operational aspects of structural integrity

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
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<td>T2.2 Material characterisation</td>
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<td>TOTAL</td>
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</table>

Comments:
- International co-operation is done in T1.2 BREDAA (Barsebäck, Ringhals)
- In WP3 (T1.2) 7 k€ will be used by Kim Wallin to contribute to two ASTM Committee E08 Fatigue and Fracture executive committee meetings in 2016, held in Toronto and Atlanta.
- In WP3 (T2.2) 3 k€ will be used by Sebastian Lindqvist to contribute to ICF14 conference in June.
MOCCA

Mitigation of cracking through advanced water chemistry

Authors Timo Saario, Konsta Sipilä, Aki Toivonen and Essi Jäppinen
Affiliation VTT Technical Research Centre of Finland
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1. Research theme and motivation

1.1 Background and state-of-the-art

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 due to impurity enrichment. Typical crevice liquid may be highly caustic with pH_{H2O} > 10 (NaOH) in addition to several other corrosive species causing pitting corrosion, denting and stress corrosion cracking.

There are three main routes to mitigate the corrosion problems caused by magnetite deposition. 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 done e.g. by controlling the secondary side water pH to be between 9.6 and 10, which coincides with the minimum in magnetite 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 TiO2 or a film forming amine).

Film forming amines (FFA) have been found 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% [1], even at elevated pH of 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 in only one PWR plant, there is a need for further studies on their application.

![Figure 1. Susceptibility to PbSCC of different SG tube materials exposed to a crevice solution of 10% NaOH at T = 315°C, (a) no additives and (b) with 1000 ppm Pb. [3]](image)

Figure 1. Susceptibility to PbSCC of different SG tube materials exposed to a crevice solution of 10% NaOH at T = 315°C, (a) no additives and (b) with 1000 ppm Pb. [3]

Lead has been detected in effectively all tubesheet samples, crevice deposits and surface scales removed from SGs. Typical concentrations are 100 to 500 ppm but in some plants, concentrations as high as 2,000 to 10,000 ppm have been detected [2]. The SG tube materials considered to be most resistive towards SCC, i.e.
Alloy 600TT, Alloy 800 and Alloy 690 all have been shown to be susceptible to SCC enhanced (in case of 690TT and 800) by presence of lead (PbSCC), Fig.1. The cracking susceptibility has a strong dependence on the redox-potential of the crevice environment. Redox-potential, on the other hand, is strongly affected by the amount of copper oxide in the crevice solution [3]. There is a clear need for clarification of the mechanism of PbSCC and the possibility of mitigating it through introduction of different inhibitors.

Hydrazine (N$_2$H$_4$) is routinely used at PWRs as an oxygen scavenger as part of the secondary side water chemistry. During power operation the concentrations are typically below 100 ppb. Preservation of SGs during outages requires much higher concentrations of hydrazine. There is a distinct possibility that, because of the health and environmental risks related to the use of hydrazine, the EU will in the future pass a directive forbidding its use. Already at the moment, the Finnish environmental regulations have been tightened so that the use of high concentrations of hydrazine during outages is becoming impossible. Alternative water chemistries need to studied, including mixtures of other amines and the use of film forming amines.

Primary water stress corrosion cracking (PWSCC) is much less common than SCC in the PWR secondary side. Apart from steam generator service failures caused by PWSCC, failures have also occurred in for instance vessel head penetrations, bottom mounted instrument nozzles and reactor pressure vessel nozzle safe-end welds, mainly in Alloy 600. Research has mainly focused on the effects of cold deformation [5] and hydrogen concentration. The effect of elevated boron (B), lithium (Li) and potassium (K) on stress corrosion susceptibility of primary circuit structural materials has so far been studied in a rather limited scope [5-7]. Boron is indicated as an accelerator of crack growth in Alloy 600 (Fig. 2). Alloy 690, which is more common in new builds, is less susceptible to PWSCC than Alloy 600 but not by far immune [8]. Plants are aiming at longer fuel cycles requiring higher B and Li/K concentrations at beginning of cycle. This project aims at increasing understanding of the possible role of B and Li/K in PWSCC crack initiation and growth.

![Figure 2. Effect of boron concentration on crack growth rate of Alloy 600 in PWR primary water. T = 330°C, K = 25MPam$^{1/2}$, [7]. The data points represent different Li-concentrations.](image)

The present study aims at developing knowledge and PWR secondary side water chemistry programs enabling minimisation of magnetite formation in the feed water line and deposition of magnetite into SGs as well as for mitigation of corrosion phenomena in SG crevices related to deposition and impurity enrichment. Alternative water chemistry programs to replace N$_2$H$_4$ are investigated. The present study also aims at clarifying the role of boron, lithium and potassium in PWSCC of stainless steels and nickel base alloys. The expected outcome will improve the knowledge basis on which decisions on advanced secondary side water chemistries are made. One specific target is to study the possibility of inhibiting PbSCC in SGs through the use of film forming amines such as octadecylamine, ODA.
1.2 Objectives and expected results

The main objectives of the present study aim at developing knowledge and PWR/WWER secondary side water chemistry programs enabling

- replacement of hydrazine in the secondary side both during outage (in SG preservation) and power generation
- minimisation of magnetite formation in the feed water line
- minimisation of deposition of magnetite particles into SGs
- mitigation of corrosion phenomena in SG crevices related to deposition and impurity enrichment

Specifically, as a result of the study the susceptibility of carbon steel to lead assisted stress corrosion cracking (PbSCC) will be clarified. The role of boron, lithium and potassium on PWSCC of stainless steels and nickel base alloys in PWR/WWER primary water is investigated. The expected outcome is to improve the knowledge basis on which decisions on advanced secondary side water chemistries are made. These results will be used in plant life extension management programs.

1.3 Exploitation of the results

Already at the moment nearly 100% of the USA PWR plants are using secondary side water chemistry programs consisting of a combination of several amines. Additional elements, such as the use of dispersants and film forming amines are currently being studied worldwide. The choice of an optimal combination is somewhat plant specific, so there is a need for deeper knowledge of the effects of different amines and other potential chemicals.

Within EU, replacing hydrazine in the secondary side water chemistry is becoming a more important issue. Alternative approaches to the use of hydrazine during outages have already become an acute research issue.

The results of this project will be exploited when considering the use of different water chemistry alternatives. The first results on film forming amines and effects of elevated pH can be applied towards the end of 2017. The results from studies of the effects of different amines and combination of amines as well as the results from mitigation of PbSCC can be applied in a longer run, starting from 2017. In both cases the end users are the plants (Loviisa 1, Loviisa 2, Olkiluoto 3 and Hanhikivi 1) and authority (STUK) in Finland as well as the nuclear community as a whole.

1.4 Appropriateness of the project to SAFIR2018 programme

The main research subjects in this project, dealing with developing alternative water chemistry programs for mitigating cracking strongly support the SAFIR 2018 target of increasing the readiness for actions aiming at better predictability of plant and component ageing management. Water chemistry programs designed to mitigate component cracking can be utilised in view of the currently operating plants, and also in relation to plants under construction or design. A specific part of the project is designed to support strengthening of international cooperation and the utilization of the results from that cooperation. In the plan, several recently graduated research scientists are being further trained to work in the nuclear field, and new research trainees (undergraduates) will also take part in the work as part of their MSc thesis work.

1.5 Education of experts

The project aims at combining knowledge regarding materials science, electrochemistry and water chemistry and as such, offers an exceptionally broad basis for new experts to be trained into the nuclear power plant safety area. The expected theses and dissertations are described below and shown also in Table 1.

Mrs. Essi Jäppinen, young scientist at VTT who graduated in 2014 started her PhD studies in 2016. The focus of the work is on developing experimental tools and understanding on deposition of corrosion products and localised corrosion in secondary circuit conditions of a PWR/WWER. She has published five conference papers and accomplished roughly 20% of the studies required. Mr. Konsta Sipilä, young scientist at VTT who graduated in 2013 has started his PhD studies in 2015. As of now, he has published three journal articles of the five articles
planned, submitted the fourth and fifth articles for publication and accomplished roughly 40% of the studies required. The focus of Mr. Sipilä’s PhD thesis work is on application of the controlled distance electrochemistry (CDE) technique in studying corrosion phenomena in both BWR and PWR/WWER environments. Both Mrs. Jäppinen and Mr. Sipilä will work as research scientists in this project. Ms. Sari Järvimäki, chemist working at Fortum Power and Heat Loviisa nuclear power plant in the chemistry laboratory group has started her PhD work in 2012. Her thesis focuses on deposition processes of soluble iron and iron containing particles in PWR/WWER secondary side environment including also the stability of oxide films forming on structural materials. She has published two journal articles of the five articles planned and accomplished roughly 33% of the studies required.

In addition to the three dissertations mentioned above, one MSc –thesis focusing on lead assisted stress corrosion cracking of steam generator tubing is planned for the duration of the project. The MSc-thesis is planned to be executed in 2018.

Table 1. Education of experts / PhD dissertations and MSc thesis.

<table>
<thead>
<tr>
<th>Researcher</th>
<th>Target</th>
<th>Title / subject area</th>
<th>Progress</th>
<th>Expected ready</th>
</tr>
</thead>
<tbody>
<tr>
<td>Konsta Sipilä</td>
<td>PhD</td>
<td>High temperature electrochemical impedance measurements in light water reactor environments</td>
<td>3 published and one submitted journal papers, 40 % of studies</td>
<td>2019</td>
</tr>
<tr>
<td>Essi Jäppinen</td>
<td>PhD</td>
<td>Magnetite deposition and localised corrosion in PWR secondary side</td>
<td>5 conference papers, 20 % of studies</td>
<td>2020</td>
</tr>
<tr>
<td>Sari Järvimäki</td>
<td>PhD</td>
<td>Oxide film stability and iron deposition in PWR secondary side</td>
<td>1 journal article, 1 conference paper, 33% of studies</td>
<td>2018</td>
</tr>
<tr>
<td>NN</td>
<td>MSc</td>
<td>Mitigation of PbSCC in SG tubing</td>
<td></td>
<td>2018</td>
</tr>
</tbody>
</table>
2. Work plan

General work plan for 2015-2018.

This project focuses on developing water chemistry knowledge enabling minimisation of corrosion damage accumulation in primary and secondary circuits of PWRs/WWERs.

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. 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. Another clearly established cause of SG corrosion damage is the lead assisted stress corrosion cracking, PbSCC. This study aims at finding suitable inhibitors preventing SCC in general and PbSCC in particular in SGs.

As part of the results, three PhD dissertations and one MSc thesis are expected to be carried out. More specifically, the following research themes will be focused on:

- possible replacement of hydrazine as oxygen scavenger in the secondary side both during power operation and outages
- the use of film forming amines, especially octadecylamine (ODA) in passivating the feed water line and minimising carbon steel corrosion rate
- the use of elevated pH (9.6 … 9.8) and its possible effects on existing deposits (especially Cu)
- studying possible benefits of using ODA as a corrosion inhibitor for Cu-containing components
- clarifying the role of boron, lithium and potassium in PWSCC of stainless steels and nickel base alloys
- the use of amine mixtures for pH control in the high pH range and possible effects on magnetite deposition
- development of research tools for simulating the SG crevices in order to be able to study experimentally localised crevice chemistry and enrichment of impurities
- mitigation of SCC in SG crevices through application of inhibitors.

The work plan for 2017 consists of the following work packages. Partners and person months allocated to the work plan in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in Work Plan 2017</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>8.4</td>
</tr>
<tr>
<td>UCTM*</td>
<td>2.0</td>
</tr>
<tr>
<td>Fortum Power and Heat**</td>
<td>1.0</td>
</tr>
<tr>
<td>Manchester University**</td>
<td>0.5</td>
</tr>
</tbody>
</table>

*subcontracted work; UCTM = University of Chemical Technology and Metallurgy, Sofia, Bulgaria
**in-kind contribution
2.1 Work package 1 (WP1), Hydrazine replacement in PWR/WWER

2.1.1 Task 1.1 (T1.1); Alternatives for hydrazine replacement

T1.1 goal in 2016 was to perform a state-of-the-art literature study on possible ways to replace hydrazine as an oxygen scavenger in the PWR/WWER secondary circuits and to prepare a plan for experimental studies to be carried out in 2017.

In 2016, a state-of-the-art report was prepared [9]. According to the study, at the moment no straightforward alternative for hydrazine in power plants is available. Thus, e.g. EdF has taken the approach to provide a hydrazine leak-proof and confined plant environment in order to minimize hydrazine contact with operating staff and hence eradicate risks to health.

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

In the study it was concluded that instead of trying to identify a single chemical which meets both of these criteria (as hydrazine does), it might be worthwhile looking for a combination of chemicals or treatments which together would fulfill both the primary requirements for an oxygen scavenger. One could for example choose to passivate the feed water line surfaces with a suitable film forming amine and to keep the oxygen level at a minimum by feeding a strong reducing agent which does not have the negative environmental and health risks that hydrazine has.

2.1.2 Task 1.2 (T1.2); Experimental verification

The effectiveness of several alternatives as oxygen scavengers is compared experimentally to that of hydrazine. The conditions are representative of a steam generator under revision period, i.e. room temperature and basic pH. Especially alternatives which are active at low temperature are of interest here.

The expected result is a VTT Research Report on the effectiveness of hydrazine alternatives as oxygen scavengers under simulated PWR secondary side revision period conditions.

Partners and person months allocated to WP1 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.6</td>
</tr>
<tr>
<td>UCTM</td>
<td>1.4</td>
</tr>
</tbody>
</table>

2.2 Work package 2 (WP2), Magnetite formation in feed water line

2.2.1 Task 2.1 (T2.1); Effect of [ODA] and T on ODA-film

T2.1 goal in 2016 is to start the study on the kinetics and effectiveness of octadecylamine (ODA) film formation on carbon steel under PWR/WWER secondary side water chemistry conditions, as a function of ODA concentration and temperature. These tests are currently underway, and are expected to increase knowledge on the effect of temperature and ODA concentration on the degree of carbon steel corrosion inhibition.
2.2.2 Task 2.2 (T2.2); Long term effectiveness of ODA preservation

This Task will be started later.

2.2.3 Task 2.3 (T2.3); Effect of potential on ODA preservation

Steam generator preservation during power operation and during shut-down are both important for the safety of the plant. Film forming amines (such as ODA) have been used to improve the passivity of the feedwater line, which then reduces the amount of magnetite carried into the steam generator thus reducing the risk for deposit formation and localised corrosion. In case of oxygen in-leakage potential within the steam generator can increase making components susceptible to stress corrosion cracking and other forms of localised corrosion.

In this Task, the effect of potential on the protectiveness of ODA-film on carbon steel is studied experimentally in the temperature range $T = 25^\circ C \ldots 300^\circ C$. A simulated oxygen in-leakage is performed by increasing the potential of a carbon steel sample previously covered with an ODA-film. The response to the potential increase is measured in situ using electrochemical methods.

This Task will be started later.

2.2.4 Task 2.4 (T2.4); Other film forming amines

This Task will be started in 2018.

2.3 Work package 3 (WP3), Magnetite deposition into SG

2.3.1 Task 3.1 (T3.1) Effect of temperature on magnetite charge

$T3.1$ goal in 2016 is to upgrade the streaming potential experimental arrangement for studying magnetite surface charge in simulated SG environment to enable measurements up to $T = 300^\circ C$ and to verify the functioning of the arrangement by a limited experimental program. This work is currently underway.

2.3.2 Task 3.2 (T3.2) Effect of ODA injection on magnetite charge

$T3.1$ goal in 2017 is to apply the streaming potential measurement system in studying the effect of ODA injection on magnetite charge. Previously performed literature study [1] indicated that ODA injection may result in partial re-entrainment of deposits into the flow, at least in cases where deposit consolidation has not taken place to a significant degree. One possible reason for this can be that ODA changes the surface charge of magnetite. Such a change can also affect the deposition behaviour of fresh magnetite particles.

The expected result is a VTT Research Report on the effect of ODA on magnetite surface charge as a function of temperature in simulated PWR secondary side environment.

Partners and person months allocated to WP3 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
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</tr>
<tr>
<td>Fortum*</td>
<td>0.5</td>
</tr>
<tr>
<td>The University of Manchester*</td>
<td>0.5</td>
</tr>
</tbody>
</table>

*in-kind contribution
2.3.3 Task 3.3 (T3.3) Effect of other FFA’s on magnetite properties

This Task will be started later.

2.4 Work package 4 (WP4), Mitigation of PbSCC

2.4.1 Task 4.1 (T4.1) Arrangement for experimental studies of PbSCC

T4.1 goal in 2016 was to construct an experimental arrangement for studying SCC in simulated SG environment and to verify the arrangement by a limited experimental program on susceptibility of low-alloyed steel (WWER SG primary collector body material) and/or Alloy 690TT to PbSCC. This work has been concluded and the report is being written. WWER SG primary collector body material, carbon steel 22K was used as the test material. The experimental set-up developed consists of a static autoclave equipped with a step-motor driven external loading system and a simultaneous arrangement for electrochemical measurements. Combination of stress corrosion cracking experiments with the slow strain rate test (SSRT) method and electrochemical methods revealed that 22K is susceptible to PbSCC in SG crevice conditions under slightly elevated potential.

2.4.2 Task 4.2 (T4.2) Mechanism of PbSCC in carbon steel and Ni-based alloys

T4.2 goal in 2017 is to study the mechanism causing PbSCC susceptibility of carbon steel 22K. Tests will be conducted in a simulated SG crevice environment at T = 278°C.

The expected result is a scientific journal publication on PbSCC susceptibility of carbon steel 22K.

Partners and person months allocated to WP4 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
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<td>UCTM</td>
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</tr>
<tr>
<td>Fortum*</td>
<td>0.5</td>
</tr>
</tbody>
</table>

*in-kind contribution

2.4.3 Task 4.3 (T4.3) Inhibitors for PbSCC of LAS and Ni-based alloys

This Task will be started later.

2.5 Work package 5 (WP5), Effects of B and Li/K on PWSCC

In 2016 VTT performed by her own funding a study on the effect of lithium concentration (2.2 and 10 ppm Li) and temperature (T = 325°C and 340°C) on the PWSCC susceptibility of sensitised Alloy 600. The results will be published as the MSc-thesis of Mr. Antti Tuhti. This work covers also the state-of-the-art study that was planned to be executed as Task 2.5.1 in the current MOCCA-project. Thus, this Task will not be performed. Further actions regarding this work package will be considered based on the mentioned MSc-thesis work.

2.5.1 Task 5.1 (T5.1) State-of-the-art study

See above, this Task is already covered. There will be no separate report on this issue since the MSc-thesis will be public.
2.5.2 Task 5.2 (T5.2) Experimental verification

This Task will be started in 2018 if the state-of-the-art study shows that there is a clear research need.

2.6 Work package 6 (WP6), International cooperation


Work package 6 focuses on strengthening the international network in the area of water chemistry of NPPs and the application of water chemistry programs for mitigation of stress corrosion cracking. The main forums for international cooperation are the Nuclear Plant Chemistry (NPC) conferences arranged biennially on even numbered years, the International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors arranged also biennially but on odd numbered years, and the meetings of the International Cooperation Group on Environmentally Assisted Cracking of Water Reactor Materials (ICG-EAC) arranged on a yearly basis. VTT is also an active member of the European Cooperative Group on Corrosion Monitoring, ECG-COMON, which e.g. organises Round Robins on corrosion monitoring technologies.

Partners and person months allocated to WP6 in 2017 are given in the table below.

<table>
<thead>
<tr>
<th>Partners in WP6</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>0.3</td>
</tr>
</tbody>
</table>

2.6.1 Task 6.1 (T6.1) ECG-COMON cooperation

Task T6.1 goal in 2017 is to participate in the activities of ECG-COMON and specifically take part in a Round Robin on application of the electrochemical impedance spectroscopy in corrosion studies. The ECG-COMON yearly meeting will take place June 12-13, 2017, in Hungary.

2.6.2 Task 6.2 (T6.2) Conferences / ICG-EAC and FFP

Task T6.2 goal in 2017 is to participate in the meeting of the International Cooperation Group on Environmentally Assisted Cracking of Water Reactor Materials (ICG-EAC) to be held May 7 to 12, 2017, in Chester, UK and in the International Conference on Film Forming Amines and Products (FFP) to be held April 4 – 6, 2017, at the KKL Luzern in Lucerne, Switzerland.
3. Deliverables 2017

The planned deliverables for 2017 are listed in the tables below.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Status</th>
<th>Indicative person months</th>
<th>Deadline</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Planned pm</td>
<td></td>
</tr>
<tr>
<td>D1.2.1</td>
<td>A VTT Research Report describing the experimental results on the effectiveness of alternative oxygen scavengers under PWR secondary side feedwater line water chemistry conditions.</td>
<td>1.6</td>
<td>15.09.2017</td>
<td></td>
</tr>
<tr>
<td>D3.2.1</td>
<td>A VTT Research Report describing experimental results on the effect of ODA on magnetite surface charge</td>
<td>2.0</td>
<td>30.11.2017</td>
<td></td>
</tr>
<tr>
<td>D4.2.1</td>
<td>A scientific publication on the PbSCC of carbon steel 22K</td>
<td>4.5</td>
<td>15.12.2017</td>
<td></td>
</tr>
</tbody>
</table>
4. Project organisation

The project manager will be DSc Timo Saario from VTT. VTT is also responsible for the whole project.

The main researchers, their organisation, the tasks they will be contributing, and the estimated person months in 2017 are shown in the table below.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PhD Martin Bojinov</td>
<td>Professor</td>
<td>UCTM</td>
<td>T1, T4</td>
<td>(2.0)*</td>
</tr>
<tr>
<td>Eng Tiina Ikäläinen</td>
<td>Engineer</td>
<td>VTT</td>
<td>T3, T4</td>
<td>1.0</td>
</tr>
<tr>
<td>MSc Essi Jäppinen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T3, T4</td>
<td>2.0</td>
</tr>
<tr>
<td>MSc Sari Järvimäki</td>
<td>Chemist</td>
<td>Fortum</td>
<td>T3, T4</td>
<td>(1)**</td>
</tr>
<tr>
<td>DSc Timo Saario</td>
<td>Principal Scientist</td>
<td>VTT</td>
<td>T1, T3, T4, T6</td>
<td>3.0</td>
</tr>
<tr>
<td>MSc Konsta Sipilä</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T1, T4, T6</td>
<td>2.0</td>
</tr>
<tr>
<td>DSc Aki Toivonen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T4</td>
<td>0.4</td>
</tr>
<tr>
<td>MSc Max Szolcek</td>
<td>Postgraduate</td>
<td>The University of Manchester</td>
<td>T3</td>
<td>(0.5)**</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td>8.4 (+2.0)* (+1+0.5)**</td>
</tr>
</tbody>
</table>

*The person-months in the parentheses include the work conducted at the University of Chemical Technology and Metallurgy (UCTM) in Sofia, Bulgaria, which will be performed as subcontracted work

**The person-months in the parentheses include the in-kind contribution of Fortum Power and Heat Loviisa NPP and the University of Manchester
5. Risk management

There are no foreseeable significant risks for the implementation of the project.
References


### Work packages and Tasks

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Volume</td>
<td>Personnel</td>
</tr>
<tr>
<td>WP1 - Hydrazine replacement in PWR/WWER</td>
<td>1.6</td>
<td>19</td>
</tr>
<tr>
<td>T1.2 Experimental verification</td>
<td>1.6</td>
<td>19</td>
</tr>
<tr>
<td>WP2 - Magnetite formation in feed water line</td>
<td>0.0</td>
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<tr>
<td>No work in 2017</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>WP3 - Magnetite deposition into SG</td>
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<td>22</td>
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<tr>
<td>T3.2 Effect of ODA injection on magnetite charge</td>
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<td>WP4 - Mitigation of PbSCC</td>
<td>4.5</td>
<td>64</td>
</tr>
<tr>
<td>T4.2 Mechanism of PbSCC in LAS and Ni-based alloys</td>
<td>4.5</td>
<td>64</td>
</tr>
<tr>
<td>WP5 - Effects of B and Li/K on PWSCC</td>
<td>0.0</td>
<td>0</td>
</tr>
<tr>
<td>No work in 2017</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>WP6 - International cooperation</td>
<td>0.3</td>
<td>4</td>
</tr>
<tr>
<td>T6.1 ECG-COMON cooperation</td>
<td>0.1</td>
<td>1</td>
</tr>
<tr>
<td>T6.2 Conferences / ICG-EAC and FFP</td>
<td>0.2</td>
<td>3</td>
</tr>
<tr>
<td>TOTAL</td>
<td>8.4</td>
<td>109</td>
</tr>
</tbody>
</table>

### Comments:

The person-months shown in the table (8.4), is the work conducted by VTT personnel. In addition there will be 2.0 person-months by prof. Martin Bojinov / UCTM (subcontracted work), 0.5 person-months by MSc Max Szolcek / Univ. of Manchester (in-kind) and 1.0 person-month as in-kind contribution by Fortum.

The travel costs are for 1) participation in the ECG-COMON -meeting, 2) participation in the ICG-EAC symposium in Chester, UK and 3) participation in the FFP conference in Lucerne, Switzerland.
SAFIR2018 Project plan 2017

THELMA

Thermal ageing and EAC research for plant life management

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1. Research theme and motivation

The project THELMA, Thermal ageing and EAC research for plant life management, deals with nuclear materials behaviour in LWR environments with special focus on determination of thermal ageing in austenitic primary circuit materials (stainless steel weld and cast materials as well as Alloy 690 and Alloy 52 weld metal), the effect of irradiation on internals, the effect of environment on fatigue life and on precursors for environmentally assisted cracking initiation to be used for plant life management and failure analyses. In 2017 a new topic is introduced, i.e., investigations on the correlation between pressure vessel steel microstructure and mechanical properties. Educating new experts in the field of nuclear materials is of high priority in the project. By the time of this proposal one thesis and six scientific publications has been produced in THELMA2015 and 2016, one thesis work is ongoing and one scientific publication and two research reports are still foreseen by the end of December 2016.

1.1 Background and state-of-the-art

Understanding and measuring the long-term effects of LWR environments on the characteristics of nuclear materials is essential for safe nuclear power plant operation. Materials are inherently subject to slow microstructural changes, i.e., thermal ageing, at LWR temperatures, and this will affect the properties of the materials. Thermal ageing changes the properties of the materials, and increases their environmentally assisted cracking susceptibility. Knowledge is not only needed on thermally aged materials, but also on material characteristics of typical nuclear components in as-manufactured condition and the influence of the operation environment, loading and irradiation to enable prediction of their behaviour during the long-term operation. THELMA will address these issues, and the knowledge is used in e.g. root cause analyses.

Thermal ageing of SSC (Systems, Structures and Components) is one of the most challenging issues of the long-term operation due to the inevitable changes in material properties in LWR environments. The as-manufactured characteristics and changes of metallic nuclear materials due to thermal ageing, irradiation, nuclear environment and loading must be known for safe operation and appropriate plant life management. Investigations on the effects of thermal ageing require aged material, which is not always possible to produce within the time frame of a SAFIR programme. Through international co-operation with Massachusetts Institute of Technology (MIT), Knolls Atomic Power Laboratory (KAPL) in US, Korean Advanced Institute of Science & Technology (KAIST) in Korea, up to 40 000 h (4.6 y) aged stainless steel weld metal, duplex stainless steel materials and nickel-based Alloy 690 are available for investigations within THELMA. Co-operation with Vattenfall, Sweden, has long traditions within the field of nuclear materials, and this has been extended in THELMA to co-operation on ageing of cast stainless steels utilising plant aged materials (up to 70 000 h). The co-operation includes in-kind results from the partners own projects on these subjects, which already as such creates results beyond an usual level. Further, the international co-operation and joint publications result in benchmarking of our scientific level and improved visibility of our results and capabilities.

In THELMA, thermal ageing of Type 316L weld metal, CF8M cast stainless steel and Alloy 690 are investigated. The main issues for these materials are described in the following, while the progress achieved in THELMA2015 and 2016 are described together with the THELMA2017 work plan.

Thermal ageing of weld metals, which comprise of austenite with some 𝛿-ferrite, is worldwide considered to be an issue requiring further studies. Weld metals are additionally subject to synergistic effects of thermal ageing and irradiation, which is an almost fully non-explored issue. Ageing of Type 316L stainless steel welds, i.e., same type as used in OL3, thermally aged up to 40 000 h at 300°C have been investigated both in an MIT-project and in THELMA. The investigations are now mostly done, and will be finalised in 2017.
PNNL and EPRI have submitted a proposal for a project dealing with the synergistic effect of thermal ageing and irradiation of stainless steel weld metals and cast stainless steels. MIT was earlier a partner and THELMA was then offered a possibility to participate. THELMA will explore the possibilities to participate in this project, which would give valuable early information on these effects.

Thermal ageing of cast stainless steels, used in Finland especially in valves and pumps, were investigated extensively in the 1980’s and 90’s. These investigations revealed spinodal decomposition as the ageing mechanism. However, recent international studies have revealed G-phase precipitation, carbide/nitride precipitation as well as segregation to play an important role in the process, which is the reason for further studies in the THELMA project. The research increases not only the understanding of thermal ageing in cast materials but is valuable also for the understanding of weld metal thermal ageing. The work is done on plant aged material received from a Swedish power plant, and is done as part of a thesis work performed at KTH and Studsvik Nuclear AB.

Thermal ageing of nickel-based materials, such as Alloy 690 and Alloy 52/152 weld metals, degrades their stress corrosion cracking resistance. The effect has been attributed to embrittlement related to the formation of an ordered Ni4Cr intermetallic phase during short- and long-range ordering reactions, or to the precipitation of carbidic during ageing. Only few investigations have been performed and the characterization of short-range ordering is difficult, but this can be an important ageing mechanism in modern NPP’s. Much more research is needed to verify the long-term behaviour of these materials and the respective influence of short- or long-range ordering (SRO and LRO) and carbide precipitation. In THELMA, thermally aged Alloy 690 with different ageing times and treatments (heat treatment and cold working) is investigated using versatile techniques, aiming at assessing the propensity for embrittlement and EAC. In the case of Alloy 52 weld metal, the thermal embrittlement of pure Alloy 52 simulates V-grove welds. Additionally, the narrow gap welds, which are diluted by the base materials, are investigated to assess their propensity to thermal ageing later in the project. The work is part of ongoing doctoral thesis work at Aalto University.

In addition to the thermal ageing, THELMA also deals with investigations on the effect of irradiation on internals. Understanding the effect of irradiation on RPV internals behaviour is essential for safe and economic plant operation. Knowing the appropriate time for component exchange has a huge impact on plant availability. It has also a huge economic impact due to high neutron activity of these components. Both mechanical and microstructural development are subject to continuous research, e.g. within the OECD Halden project. VTT has for many years characterised the materials used in these investigations. These will be continued in THELMA2017.

Assessment of the fatigue resistance under operating conditions is required in the YVL-guide E.3 for the approval of materials used in piping in safety class 1. The justification procedure can be chosen by the applicant. A commonly used guidance comes from NRC RG1.207, given by the NRC, US, while various national programmes in Europe aim to develop counter proposals allowing greater operational efficiency with at least comparable safety assurance. The data obtained in laboratory fatigue tests, and used, e.g., as a basis for the revised ASME fatigue curve, do not reflect accurately in-plant observations. The lack of correlation between the laboratory test data and the in-plant operating experience compromises somewhat the confidence in corrosion fatigue assessment in light water reactor (LWR) environments, thus impeding total safety management of the NPPs from being developed. The 5 year EU H2020 project proposal INCEFA PLUS, INcreasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment, brings these programmes together through which a strong EU response to the NRC methodology will be obtained with improved safety assurance through increased lifetime assessment reliability. The goal is a reduced assessment uncertainty enabling easier maintenance of safety. Although current EU rules do not accept formal combination of national programs and EU-project, THELMA will act as the national information exchange platform for the INCEFA PLUS project. The project consortium has agreed to share the results to the project partner national interest groups.

Environmentally assisted cracking (EAC) continues to influence plant performance and availability significantly. Recent examples are EAC in Ni-base alloys in pressurised water reactors (PWRs), in cold-worked stainless steel (SS) and dissimilar metal welds in PWRs and boiling water reactors (BWRs), as well as irradiation-assisted SCC incidents in both types of reactors. Reliable prediction of the EAC behaviour, i.e., both crack growth rates and initiation susceptibility of given materials is considered very important for safe long-term operation of plants. Initiation is a complex phenomenon with numerous precursor events, some of which originate already from manufacturing. Determination, especially by statistically valid measurements, of initiation is very demanding and time consuming. Development of a validated accelerated test method for EAC initiation would increase the
possibilities to develop predictions for EAC crack initiation and understanding of the role of different precursors. A joint, in-kind project, MICRIN PLUS (Mitigation of CRack Initiation PLUS) was launched in 2015 under the NUGENIA+ project to tackle this issue. The main objective of the MICRIN+ project was to develop a first draft of a “NUGENIA proposal for optimized surface conditions to mitigate in-service degradation” for implementation in general codes & standards. Also MICRIN+ participants have agreed to share the results to the national interest groups, and THELMA will be the forum for this. The MICRIN+ project was finalised in 2016, and further work is planned to be performed in the MEACTOS project, which was submitted to the EU-H2020 in 2016. A decision on future initiation testing in THELMA will be made when the results of the H2020 proposals are available.

The structural integrity of the pressure vessel is the most important issue for any nuclear power plant. The assessment is based on surveillance programs while additional research is continuously performed for an improved mechanistic understanding. These investigations have, e.g., revealed that irradiation causes formation of agglomerates, which cause embrittlement of the RPV material. A number of issues remains, though, open and represent knowledge gaps with respect to the irradiation-induced ageing of low alloy steels and the effect on the reactor pressure vessel and their fitness for long-term operation. Among these issues are representativeness of the production weld test blocks that make up the surveillance programs to the actual pressure vessel welds, the effect of macroscopic weld material inhomogeneity on mechanical behaviour measured as either fracture toughness or impact toughness and the attenuation, i.e., the decreasing of the irradiation-induced defect number with increasing depth from the media touched surface of the RPV, the role of flux, etc. By the closure of the nuclear units at the Barsebäck site, a unique opportunity has opened up to harvest samples from the reactor pressure vessels and comparing the material properties with those of the surveillance materials. The BREDA-project (Barsebäck REsearch&Development Arena) is relevant also for Finland. The OL1 and OL2 RPV weld metal is of the same type as the Barsebäck weld, and manufactured by the same company, Uddcomb. The chemical composition is a high Ni, high Mn steel, which is known to be prone to embrittlement. The FH1 weld metal will obviously also be a high Ni material (although with higher Cr-content). The work will be performed in co-operation with the SAFIR2018 LOST project, a doctoral thesis work at KTH (Magnus Boåsen) and with a consortium consisting of Vattenfall (Ringhals and Forsmark NPP’s), TVO, Fortum, Fennovoima and Energiforsk (Sweden). The work in THELMA will focus on characterisation of the microstructure from macro- to nano-scale to increase the connection between the mechanical behaviour and the microstructure.

International co-operation has always been strong in the field of nuclear materials. Such co-operation is crucial for getting access to the most recent international research results and plant operation experiences, to create networks, and to benchmark the scientific level of our research. Knowledge transfer and education of new experts is a continuous process as part of the every-day work life. However, also more structured knowledge transfer is needed, e.g. in the form of courses, workshops and accessibility to reports. THELMA continues to update the digital archive on VTT-reports on nuclear materials. THELMA results will be published at main conferences in the field as well as scientific publications in journals. The conferences act also as excellent teaching and networking forums.

1.2 Objectives and expected results

The objectives of THELMA are to understand the underlying mechanisms and effects of thermal ageing, irradiation and nuclear environment on austenitic nuclear materials, i.e., austenitic stainless steel weld metals, cast stainless steel CF8M and nickel-based material Alloy 690. In 2017, investigations are started also on the effect of nuclear environment on pressure vessel steel weld metal with the objective of producing data on the correlation between microstructure and mechanical properties. The objective of investigation method development is to constantly improve our capabilities utilised in root cause analyses for our licensees.

The expected results of the THELMA project are:
- Determination of the activation energy for spinodal decomposition and G-phase formation in Type 316L weld metal and comparison to that of cast CF8M stainless steel.
- Determination of the changes in properties due to short-range ordering in Alloy 690.
- First steps towards best practises for surface quality in primary components.
- New experimental data and new guidelines for assessment of environmental fatigue damage to ensure safe operation of European nuclear power plants.
- Benchmarking of our capabilities to perform initiation testing in simulated LWR conditions.
Data on the correlation between mechanical and microstructural characteristics of pressure vessel materials.

Education of new nuclear materials experts.

Strengthened international co-operation and joint scientific publications.

### 1.3 Exploitation of the results

The results will be exploited by VTT in root cause analyses performed for the licensees and in expert statements on materials behaviour in nuclear environments for both STUK and the licensees. Building up knowledge for failure analyses is extremely important, as most failure analyses are to be made within a very short time period during the outages, without any possibilities to add basic knowledge during the course of those assignments. Further, the results shall be exploited by the licensees for their lifetime management and by STUK in their work on securing safe nuclear power production in Finland. The results can also be utilised in regulation harmonisation work ongoing in Europe. Most of the results can be used within 1-3 years, but will not be outdated in this time frame, though.

### 1.4 Appropriateness of the project to SAFIR2018 programme

The objectives of the THELMA project are very well in line with the objectives of the SAFIR2018 programme. It answers to the research need on ageing of metallic materials, both old and new, on the need for international co-operation, utilisation of material from decommissioned NPP’s, etc. The most important statements in the SAFIR2018 framework programme, relevant to THELMA, are collated in Table 1 together with comments on how THELMA is related to it.

Potential safety issues connected to load following operation are highlighted in the update of the framework to the 2017 call. Load following will increase the amount of transients, which are known to be one of the main factor for environmentally assisted crack initiation. It will also increase EAC crack growth. These aspects can be investigated using the same methods for crack initiation as used in THELMA2016 in the MICRIN+ project, including transients. However, autoclave investigations are tedious and expensive, and to obtain quantitative data, a much bigger, multi-laboratory programme would be needed. These aspects are therefore not included in the THELMA2017 proposal, but can be included in THELMA2018 if the MEACTOS project, which would support such investigations quite well, is approved. INCEFA+ supports the question on load following consequences.

Table 1 Summary of main items, described in the SAFIRIR208 framework programme, relevant for THELMA and corresponding THELMA project work package with a short comment.

<table>
<thead>
<tr>
<th>SAFIR2018 framework programme</th>
<th>THELMA proposal, Work package, Task and comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>p12: 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.</td>
<td>WP1-WP4. THELMA meets the SAFIR2018 vision. The PM is an internationally well recognised expert in her field, the project has a very strong international co-operation scheme and the scientific level is high. The new top of the line equipment in the Centre for Nuclear Safety is used in the research.</td>
</tr>
<tr>
<td>p21: International co-operation is a necessity in the nuclear power field, particularly for small nuclear power countries such as Finland. P49: Connecting the research with international collaboration is justified from several perspectives….</td>
<td>THELMA has international co-operation in all WP’s.</td>
</tr>
<tr>
<td>p28: Another topical issue is the utilisation of the decommissioned Barsebäck reactor in Sweden for international research projects before its dismantling in 2020.</td>
<td>WP3: RPV microstructure. The WP will utilise material from the Barsebäck surveillance program in 2017 and trepans removed from the plant in future years.</td>
</tr>
<tr>
<td>p29: Samples collected from power reactors are being radiated (in the Halden reactor), which will provide information on possible material aging phenomena resulting from long-term use.</td>
<td>WP3, T1.5 PI characterisation of internal materials. Irradiated stainless steels materials are characterised in the THELMA and results reported to the Halden programme.</td>
</tr>
</tbody>
</table>
A joint goal in all research is ensuring the reliability of aging management throughout the plant’s life cycle. The research area involves the aging of the plants and their components. It will become a key area when modernising them and building new ones.

Failure mechanisms and ensuring structural integrity, including:
- Material performance and aging in power plant conditions

The research area involves the aging of the plants and their components. It will become a key area when modernising them and building new ones.

Understanding and determining failure mechanisms and material performance is the main objective of THELMA.

THELMA PM is member of the ICG-EAC steering group, participates in the yearly meeting with presentations and delivers a travel report.

THELMA participated in MCIRIN+, devoted to EAC crack initiation methodology development and participates in a Round Robin exercise on EAC initiation determination.

THELMA follows the development of a project proposal on the synergistic effect of irradiation and thermal ageing on cast SS and SS weld metals.

THELMA is done by the YG (learning by doing) and mentoring is done on a daily basis.

Potential safety issues related to the operation of nuclear power plants in the load following mode are an important topic for research.

The on-going thesis work on thermal ageing of Alloy 690 will be finalised in 2017, i.e., Roman Mouginot: “Long-term stability of Ni-based Alloy 690 in modern pressurised water reactors”. The THELMA project team consists of about eight persons, with an appropriate combination of young and experienced researchers. The main part of the studies in THELMA is done by young experts, under the mentorship of more experienced scientists.

### 1.5 Education of experts

Three doctoral theses, started in SAFIR 2014 ENVIS and TEKES-SINI projects, are foreseen to be finalised within THELMA in co-operation with the TEKES NIWEL project. One thesis was finalised in 2015 and one in 2016, i.e., Matias Ahonen “Effect of microstructure on low temperature hydrogen induced cracking behaviour of nickel-based alloy weld metals” and Teemu Sarikka “Effect of strength mismatch on fracture mechanical behaviour of ferritic-austenitic interface of Ni-base dissimilar metal welds”. The on-going thesis work on thermal ageing of Alloy 690 will be finalised in 2017, i.e., Roman Mouginot: “Long-term stability of Ni-based Alloy 690 in modern pressurised water reactors”. The THELMA project team consists of about eight persons, with an appropriate combination of young and experienced researchers. The main part of the studies in THELMA is done by young experts, under the mentorship of more experienced scientists.
2. Work plan

The work plan for the THELMA2017 comprises of four work packages (WPs), as shown in Figure 1.

Figure 1 Schematic presentation of the project structure.

The first WP with four tasks deals mainly with the thermal ageing (and effect of irradiation) of nuclear materials, i.e., Type 316L weld metals (Task 1.1), cast stainless steels (SS) (Task 1.2), Ni-based Alloy 690 (Task 1.3) and irradiated stainless steel (Task 1.4). The results remarkably improve the understanding of thermal ageing mechanisms of nuclear materials as well as demonstrate the changes in material properties due to thermal ageing embrittlement. The last task in WP1 deals with the effect of irradiation on stainless steel internal materials and will benefit the OECD Halden project and the understanding of the effect of irradiation on internals.

The second WP (WP2) with two tasks deals with precursors for cracking. The first task forms the forum for informing the national interest group on the progress of a European project, i.e., INCEFA PLUS, with overall goal to create new guidelines for assessment of corrosion fatigue damage in Europe, which improves safe nuclear power plant operation. The second task (T1.2) is a benchmarking exercise for initiation testing in simulated LWR conditions.

The third WP (WP3) is new for THELMA2017, and is a starting point for a more solid combination of the mechanical and microstructural knowledge of pressure vessel weld metal embrittlement behaviour. The work is performed in co-operation with the SAFIR2018 LOST project and the BREDA project.

The forth WP (WP4) comprises of knowledge transfer and international co-operation. The knowledge transfer is included in the THELMA project work through mentoring the young project members. The international co-
The thermal ageing of three nuclear materials, i.e., Type 316L weld metals (Task 1.1), cast stainless steels (SS) (Task 1.2), Ni-based Alloy 690 (Task 1.3) is investigated in WP1. Additionally, the effect of neutron irradiation on stainless steels is investigation in Task 1.4. The progress of each task during 2015 and 2016 is summarised before the description of the work planned for 2017. Partners and person months allocated to WP1 are given in Table 3.

### 2.1 Thermal ageing of SSC materials (WP1)

The thermal ageing of three nuclear materials, i.e., Type 316L weld metals (Task 1.1), cast stainless steels (SS) (Task 1.2), Ni-based Alloy 690 (Task 1.3) is investigated in WP1. Additionally, the effect of neutron irradiation on stainless steels is investigation in Task 1.4. The progress of each task during 2015 and 2016 is summarised before the description of the work planned for 2017. Partners and person months allocated to WP1 are given in Table 3.

### Table 3: Partners and person months allocated to WP1

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>6</td>
</tr>
<tr>
<td>Aalto</td>
<td>4</td>
</tr>
<tr>
<td>Total</td>
<td>10</td>
</tr>
</tbody>
</table>
2.1.1 Thermal ageing of stainless steel weld metals (T1.1)

The thermally aged Type 316L weld metals from MIT, comprehensively investigated in SAFIR 2014 ENVIS and at MIT in a separate project, have been subjected to FEG-TEM investigations, which were not performed in any of the previous projects, in THELMA 2015 and 2016. The set of available materials is seen in Table 4. Additionally, weld metal from Sweden after 70 000h plant ageing is available.

The performed investigations on the MIT weld metals have shown an increase in the crack growth rate in BWR-conditions by a factor of two, reduced fracture toughness in BWR environment compared to that in air and increased nano-hardness of the δ-ferrite phase with increasing thermal ageing. Further, FEG-TEM investigations have shown clear indications of thermal ageing, i.e. mottled structure and G-phase formation in the δ-ferrite in the material aged at 430 °C for 10 000 h, while such indications were not observed in the material with modest ageing (300 °C/40 000 h). The material shows a consistent evolution of the ageing after DL-EPR measurements with a reactivation peak seen after ageing at 400 and 430°C. The results were reported at a conference in 2016.

Table 4: Summary of available thermally aged Type 316L materials, showing the materials, the ageing temperature and ageing times (in hours).

<table>
<thead>
<tr>
<th>Material/ageing</th>
<th>300 °C</th>
<th>400 °C</th>
<th>430 °C</th>
</tr>
</thead>
<tbody>
<tr>
<td>316L, low ferrite</td>
<td>5 000 h</td>
<td>1 000 h</td>
<td>1 000 h</td>
</tr>
<tr>
<td></td>
<td>20 000 h</td>
<td>5 000 h</td>
<td>5 000 h</td>
</tr>
<tr>
<td></td>
<td>40 000 h</td>
<td>10 000 h</td>
<td>10 000 h</td>
</tr>
<tr>
<td>316L high-ferrite</td>
<td>5 000 h</td>
<td>1 000 h</td>
<td>1 000 h</td>
</tr>
<tr>
<td></td>
<td>20 000 h</td>
<td>5 000 h</td>
<td>5 000 h</td>
</tr>
<tr>
<td></td>
<td>40 000 h</td>
<td>10 000 h</td>
<td>10 000 h</td>
</tr>
</tbody>
</table>

In 2017, FEG-TEM, nano-hardness and DL-EPR investigations will be performed on the weld metal received from Vattenfall, Sweden, and the results on both MIT and Swedish materials will be reported in a scientific publication.

2.1.2 Thermal ageing of cast stainless steels (T1.2)

The goal of the work is to characterise thermally aged CF8M material using especially FEG-TEM, to compare the results to those from weld metals to determine the role of the different ageing mechanisms to the behaviour of austenitic-ferritic stainless steels in co-operation with postgraduate student Martin Bjurman and Prof. Pål Efsing, KTH.

The work was started in THELMA2016 with FEG-TEM investigations on thermally aged CF8M cast stainless steel from hot and cold leg pipe bends, which have plant aged for 70 000 h at the Ringhals plant. Although APT investigations performed by KTH revealed G-phase formation in the hot leg material and clustering of G-phase elements in the cold leg material, G-phase were not observed in the performed FEG-TEM investigations performed in THELMA. The mechanical tests performed at KTH/Studsvik showed a decrease in the upper shelf impact energy. The nano-hardness measurements (performed in THELMA) show an about 40% increase of the δ-ferrite nano-hardness.

In THELMA2017, the materials will be investigated using the new FEG-TEM equipment with higher resolution than the one used in 2016. DL-EPR tests will also be performed and the results will be published in a scientific publication. The investigations at KTH will include fracture toughness tests, tensile testing using different strain rates and instrumented impact tests with different notch shapes. Joint meeting(s) with Martin Bjurman and Pål Efsing will be held to share results and discuss the implications of them.

2.1.3 Thermal ageing of Ni-based Alloy 690 (T1.3)

The goal of the work is to determine the characteristics and properties of thermally aged Alloy 690 materials and determine the thermal ageing mechanisms in this material.
Thermally aged Alloy 690 materials with different ageing times and treatments are available for investigations, see Table 4. The investigations, including micro- and nano-hardness measurements, FEG-SEM/EBSD/TEM, transmission Kikuchi diffraction (TKD), X-ray diffraction (XRD), mechanical spectroscopy (internal friction), thermal desorption spectroscopy (TDS) and tensile testing using digital image correlation were started in 2015 and continue in 2016-2017, when the doctoral thesis, including these results, is presented. In 2015 and 2016, the materials have been investigated using nano-indentation, Vickers microhardness, atom force microscopy, FEG-SEM, EBSD, FEG-TEM and X-ray diffraction (XRD). A good correlation between nanohardness and lattice parameter measurements indicates that SRO occurred at temperatures as low as 350 °C, with a peak at 420 °C. The results show that the critical temperature for short-range ordering depends on the chemical composition (the material investigated here has a Fe-content of 9.8 wt-%), that heat treatment promotes ordering prior to ageing, that cold work promotes fast short-range ordering in the first stage of ageing but leads to recrystallization after ageing at higher temperatures.

In 2017, further investigations on thermally aged Alloy 690 materials are performed focussing on the mechanical testing, investigations on the effect of hydrogen and filling in missing data points in the test matrix, especially with TEM characterization to investigate carbide precipitation. Investigations of thermally aged Alloy 52 weld metal using a selection of the above mentioned techniques on material from the TEKES-NIWEL project, aged at 410°C for 10 000 hours, will be started. Two additional journal articles will be submitted in 2017 for publication. The dissertation manuscript will also be written and the thesis will be defended for a doctoral degree.

Table 5: Summary of available thermally aged Alloy 690 materials, showing the material conditions, the ageing temperature and ageing times (in hours)

<table>
<thead>
<tr>
<th>Material/ageing T</th>
<th>350°C</th>
<th>420°C</th>
<th>475°C</th>
<th>550°C</th>
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<tbody>
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<td></td>
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<td>10 000</td>
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<tr>
<td>SA+20%CW</td>
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<td>3000</td>
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</tr>
<tr>
<td></td>
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</tr>
<tr>
<td>TT+20%CW</td>
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</tr>
<tr>
<td></td>
<td>10 000</td>
<td>10 000</td>
<td>10 000</td>
<td>10 000</td>
</tr>
</tbody>
</table>

2.1.4 Ageing of internal materials (T1.4)

VTT has for several years characterised stainless steel that has been used for in-pile testing in the OECD Halden reactor. Such investigations were also performed in 2015, but as a separate project due to rejection of the IRMA project, in which the Halden project was included. In THELMA2016 stainless steel from an in-core creep test was investigated to possibly find the reason for the observed shrinkage of the material in the beginning of the test. The material was characterised using FEG-TEM, but no such features, which could explain a contraction of the material were found.

In 2017, microstructural characterization of 4 dpa CW 316 Ti-stabilised austenitic SS supplied as sheet sample (HP1-9) and material cutting and further analyses from tested CT specimens of 5.9 dpa 304 L SS (CT1) and 9 dpa CW 316 SS (CT3) (IFA718) will be performed. The results are reported in conferences and/or as VTT research reports.

2.2 Environmentally assisted cracking precursors (WP2)

WP2 comprises of two tasks dealing with precursors for cracking. The first task (T2.1) forms the forum for informing the national interest group on the progress of European projects, i.e., INCEFA PLUS, with overall goal to create new guidelines for assessment of corrosion fatigue damage in Europe which improves safe nuclear power plant operation. The second task (T2.2) is a benchmarking exercise for initiation testing in simulated LWR conditions. The progress of each task during 2015 and 2016 is summarised before the description of the work planned for 2017. Partners and person months allocated to WP1 are given in Table 6.
Table 6: Partners and person months allocated to WP2

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
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</thead>
<tbody>
<tr>
<td>VTT</td>
<td>3.0</td>
</tr>
<tr>
<td>Aalto</td>
<td>0</td>
</tr>
<tr>
<td>Total</td>
<td>3.0</td>
</tr>
</tbody>
</table>

2.2.1 Corrosion fatigue assessment development (T2.1)

Task 1 will act as the forum for information exchange concerning the EU-INFECA PLUS project. The overall goal of the EU-INCEFA PLUS is to create new guidelines for assessment of corrosion fatigue damage in Europe to improve safe nuclear power plant operation. The materials comprise of one common material, i.e., nuclear grade Type 304 steel, delivered by EDF. The EU-INCEFA PLUS partners will perform tests also on other materials which are in their interest. These will include Ti- and Nb-stabilised stainless steels (German and Czech interests), and Type 316 stainless steel. VTT is responsible for the task on microstructural characterisation. The kick-off was held in June 2015, and a progress report was written for THELMA in 2015 and will also be written in 2016.

The overall goal of the EU-INCEFA PLUS and WP2 Task 1 is to create new guidelines for the assessment of corrosion fatigue damage in Europe to improve safe nuclear power plant operation. The focus of the EU-INCEFA PLUS project is on PWR conditions at elevated temperature (~300°C). At a later stage, some tests may be also carried out at lower temperatures (~200°C) which occur during transients. The parameters to be studied are hold time, mean strain and surface finish.

The kick-off for the project was held in June 2015. The main activities so far have been preparation of specimens for testing. The first batch of specimens has now been delivered to the partners and testing is ongoing in most laboratories. The second batch is foreseen in the near future. VTT has characterised available materials, and found the common Type 304 material to be inhomogeneous in grain size. Surface roughness measurements are ongoing.

In 2017, further testing materials will be characterised at VTT and fatigue testing in LWR environments will continue. A progress report on 2017 activities will be written within THELMA2017.

2.2.2 Round Robin on SCC initiation in Alloy 600 (T2.2)

One of the goals of the ICG-EAC group (International Co-operative Group on Environmentally Assisted Cracking) is to improve the testing quality performed by the members. A Round Robin (RR) on crack initiation testing was initiated by the group in 2016 as part of this effort. By participating in this RR, VTT will benchmark our capabilities to perform crack initiation testing. Additionally, we will get access to all data produced in the RR. The RR is led by AMEC, UK, while GE and PNNL (both from US) will provide Alloy 600 materials, which have been well characterised. A core group has written detailed testing procedure to which the participating partners shall comply.

In 2016, one initiation test was performed on cold-forged Alloy 600 from PNNL. The testing was performed at 350°C in simulated PWR conditions (2 ppm Li, 25 cc H₂), using a tensile specimen with pre-defined dimensions. The test was completed after 2000 hours at 90% load of the 0.2% proof stress, without any observed crack initiation, which is in line with results from other laboratories. A summary of the progress in 2016 will be written.

In 2017, one initiation test will be performed on Alloy 600 material from GE. The results will be reported at the ICG-EAC meeting in May 2017, when VTT will also get access to the results of the other participating laboratories. A final report on all results will be written for THELMA2017.
2.3 Microstructural characterisation of pressure vessel steels (WP3)

The objective of this new work package, WP3, with one task (T3.1) is to start microstructural characterisation of pressure vessel steel materials to improve the joining of knowledge of mechanical and microstructural characteristics. Partners and person months allocated to WP1 are given in Table 7.

Table 7: Partners and person months allocated to WP3

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
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<tbody>
<tr>
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<td>Aalto</td>
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<td>Total</td>
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2.3.1 Microstructural characterisation of pressure vessel steels (T3.1)

The main overall objective of the BREDA project, dealing with investigations on material from surveillance material from Barsebäck NPP and on material to be removed from the pressure vessel, is to compare these materials. The whole project is much larger than what is included in SAFIR2018 LOST and THELMA. A NKS-proposal has been submitted by P. Efsing/KTH for a joint project between KTH, Chalmers University and VTT. THELMA would get 20% of the VTT NKS funding.

The work in 2017 will consist of microstructural characterisation of reference material from the Barsebäck surveillance material. This is a preparation for testing and characterisation of material from accelerated surveillance capsules and the reactor pressure vessel itself, which will be available in 2018. The work is performed in collaboration with the SAFIR2018 LOST project, in which the basic fractographic investigations will be performed under the advice of THELMA PM, who has decade-long experience of identifying primary initiation sites and fracture modes of pressure vessel steels.

The work in THELMA2017 will focus on characterisation of cross-sections, prepared as close as possible to the primary initiation site to reveal the type of microstructure where initiation has occurred. The results and partly also the work will be shared with post-graduate student Magnus Boåsen, who is working on his thesis on modelling RPV steel embrittlement. A progress report will be written on the results obtained in 2017.

2.4 International co-operation and knowledge transfer (WP4)

International co-operation is essential to bring the latest knowledge to Finland. Accepted participation in international projects is also a proof of high scientific level of the work done in THELMA. Active participation and publication of THELMA research results at conferences and in scientific publications raises the visibility of the performed research. The work within THELMA is mainly performed by young scientists and technicians, which will thus get education in nuclear materials ageing issues under the mentorship of experienced scientists and professors. Partners and person months allocated to WP1 are given in Table 8. Professor Hannu Hänninen contributes with in-kind mentoring in this task.

Table 8: Partners and person months allocated to WP4

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
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</thead>
<tbody>
<tr>
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<td>Aalto</td>
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</tbody>
</table>
2.4.1 International co-operation and knowledge transfer (T4.1)

The goal of this task is to mentor the young project members and actively participate in international conferences, work groups and projects and bring the latest knowledge to Finland.

In 2015, THELMA participated in the 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 9-13, 2015, Ottawa, Canada, with three papers on THELMA results. Further, THELMA participated in the International Co-operative Group on Environmentally Assisted Cracking, ICG-EAC, in which the PM is a member of the board. A new pre-meeting training day was introduced in 2015, and U. Ehrnsten gave an invited tutorial lecture. A presentation on the synergy between SAFIR2018 RG5 projects and Nugenia project were given at a Nugenia TA4 meeting. In 2016, THELMA2016 attended the ICG-EAC 2016 meeting. An advanced course on nuclear materials was arranged by Prof. Hannu Hänninen, Aalto University, with about 50 participants. Lectures were given by persons participating in the THELMA project. The project did not participate in the costs for arranging the course, but gave possibilities for young scientists to participate in the course, and for VTT lecturers to prepare their presentations. The course has so far been arranged every second or third year.

In 2017, participation in the 18th Environmental Degradation conference and the ICG-EAC 2017 meeting is foreseen, presenting results from THELMA and writing travel reports. The PM is a member of the scientific board for the conferences, which require review of abstracts and papers for the conference. This work is, however, not a burden for the THELMA project as the work is mainly done during the summer holidays.

In the original application, participation of THELMA2017 YG project team members at the Aalto University VVER1200 seminar, with lecturers from Rosatom, was planned. The seminar will take place 14-16th of April. However, in accordance with the recommendation of the SAFIR2018 SG3 group, that training is not funded by SAFIR2018, this sub-task is left out from THELMA2017.

U. Ehrnsten will give an invited key-note lecture on EAC in stainless steels at the Eurocorr 2017 conference, in a special work-shop celebrating the 50th anniversary of the European Federation of Corrosion, EFCC, Nuclear Corrosion Working Party which will be arranged in Prague, September 3-10, 2017. The talk will, e.g., highlight the importance of research programmes like SAFIR in understanding and mitigating stress corrosion cracking in nuclear power plants.

Updating of the VTT digital archive (the YVL database) for nuclear material reports, developed within the SAFIR 2010 DEFSPEED project, was continued within THELMA2015 and 2016. The main focus in the THELMA project is to localise old reports still missing from the archive. All new VTT reports are added to the database by VTT archive personnel. A decision to extend the database to include all nuclear reports has been made at VTT in 2014 and numerous reports were added to the database in 2015 and 2016 as part of an internal VTT project. The work is now mainly done, and new reports are added to the archive as part of the VTT archive system, and archiving is no longer the responsibility of THELMA. If old missing reports are identified, these will be added to the archive.
3. Deliverables and milestones 2017

The THELMA2017 project has 9 deliverables, Table 9, of which one is also a milestone, i.e., a thesis ready and approved.

Table 9: Deliverables for THELMA2017

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
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<td>D1.1.1</td>
<td>Scientific publication submitted on thermal ageing of stainless steel weld and cast material</td>
<td>1.3</td>
<td>30.8.2017</td>
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<tr>
<td>D1.2.1</td>
<td>Conference publication (with peer review) on thermal ageing of cast stainless steels</td>
<td>2.3</td>
<td>30.7.2017</td>
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<tr>
<td>D1.3.1</td>
<td>Scientific publication submitted on results from investigations on thermally aged Alloy 690</td>
<td>4.0</td>
<td>30.8.2017</td>
</tr>
<tr>
<td>D1.3.2 M1.3.1</td>
<td>Dissertation approved (Roman Mougnot)</td>
<td></td>
<td>30.11.2017</td>
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<tr>
<td>D1.4.1</td>
<td>VTT Research Report on PI characterisation of irradiated stainless steel material</td>
<td>2.9</td>
<td>30.11.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Progress report on EU-INCEFA project</td>
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<td>30.11.2017</td>
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<tr>
<td>D2.2.1</td>
<td>Final report on Alloy 600 Round Robin results</td>
<td>2.3</td>
<td>30.11.2017</td>
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<tr>
<td>D3.1.1</td>
<td>VTT report on results from microstructural characterisation on RPV steel</td>
<td>1.9</td>
<td>15.11.2017</td>
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<tr>
<td>D4.1.1</td>
<td>Presentation at ICG-EAC meeting and Env. Deg. conference, key-note at Eurocorr 2017 and travel reports</td>
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<tr>
<td><strong>Total</strong></td>
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4. Project organisation

THELMA is a joint project between VTT Technical Research Centre of Technology Ltd, Nuclear Safety and Aalto University, School of Engineering, Engineering Materials. VTT is responsible for the whole project with Principal Scientist Ulla Ehrnstén acting as the project manager and Research Scientist Caitlin Hurley as the deputy project manager. Dr Risto Ilola is the project manager at Aalto University.

The partners in the joint activities within THELMA2017 are presented in Table 1 and Table 11.

Table 10: Main THELMA project team members

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
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<tbody>
<tr>
<td>Ulla Ehrnstén</td>
<td>Principal Scientist, Project manager</td>
<td>VTT</td>
<td>WP1-WP4</td>
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<tr>
<td>Mykola Ivanchenko</td>
<td>Dr.Tech (YG)</td>
<td>VTT</td>
<td>WP1</td>
<td>3.7</td>
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<tr>
<td>Aki Toivonen</td>
<td>Dr.Tech</td>
<td>VTT</td>
<td>T2.2</td>
<td>2.1</td>
</tr>
<tr>
<td>Caitlin Hurley</td>
<td>Dr.Tech (YG) Deputy PM</td>
<td>VTT</td>
<td>T2.1</td>
<td>2.5</td>
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<tr>
<td>Jari Lydman</td>
<td>M.Sc</td>
<td>VTT</td>
<td>T3.1</td>
<td>0.8</td>
</tr>
<tr>
<td>Marketta Mattila</td>
<td>Research Technician</td>
<td>VTT</td>
<td>T1.1, T1.2, T2.1, T2.2</td>
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<tr>
<td>Arto Kukkonen</td>
<td>Research Technician</td>
<td>VTT</td>
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<td>Risto Ilola</td>
<td>Dr.Tech Aalto PM</td>
<td>Aalto</td>
<td>WP1-WP3</td>
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<tr>
<td>Roman Mouginot</td>
<td>MSc (YG, doctoral student)</td>
<td>Aalto</td>
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<td>3.6</td>
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<tr>
<td>Hannu Hänninen*</td>
<td>Professor</td>
<td>Aalto</td>
<td>WP1-WP4</td>
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*Mentoring as in-kind

Table 11: Partners in joint activities within THELMA2017

<table>
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<tr>
<th>THELMA Work Plan</th>
<th>Partner</th>
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<tbody>
<tr>
<td>WP1, T1.1 Thermal ageing of SS weld metals</td>
<td>Professor Ron Ballinger, MIT, USA</td>
</tr>
<tr>
<td>WP1, T1.2 Thermal ageing of cast SS</td>
<td>MSc Martin Bjurman, Studsvik, KTH, Sweden</td>
</tr>
<tr>
<td></td>
<td>Dr.Tech, professor Pål Efsing, KTH / Ringhals, Sweden</td>
</tr>
<tr>
<td>WP1, T1.3 Thermal ageing of Ni-based Alloy 690</td>
<td>Dr.Tech Young Suk Kim, KAIST Korea</td>
</tr>
<tr>
<td></td>
<td>MSc Paavo Rautala, TVO (NIWEL project manager)</td>
</tr>
<tr>
<td>WP1, T1.4 PI characterisation of internal materials</td>
<td>Dr.Tech Torill Karlsen, OECD Halden project</td>
</tr>
<tr>
<td>WP2 T2.1 Precursors for corrosion fatigue</td>
<td>Dr.Tech Kevin Mottorshead, AMEC FW (EU-INCEFA project manager) and INCFEA project partners</td>
</tr>
<tr>
<td>WP2, T2.2 RR on SCC initiation in Alloy 600</td>
<td>Dr John Stairmaid, AMEC FW, UK, leading the RR</td>
</tr>
<tr>
<td>WP3, T3.1 RPV microstructure</td>
<td>Dr.Tech, professor Pål Efsing, KTH / Ringhals, Sweden</td>
</tr>
<tr>
<td></td>
<td>MSc Magnus Boåsen, KTH</td>
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<td>BREDA project consortium</td>
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<td>SAFIR2018 LOST</td>
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5. Risk management

The risk expressed in the THELMA2015 application on combination of SAFIR2018 projects to EU-projects, without final agreement on the administration was realized, and the projects EU-INCEFA PLUS and MICRIN+ had to be economically separated from the SAFIR2018 THELMA project, at high expenses for VTT. However, THELMA will continue to be the national forum for information exchange of these projects, which was achieved prior the final approval of these projects.

According to recent news, the Halden test reactor is shut down in mid-October for an undefined time due to economic reasons and a part of the staff is temporarily dismissed. This should, however, not affect the Halden in-kind work on characterisation of internals in THELMA2017, as the material is already at VTT. Whether the situation at Halden will affect the in-kind funding is not known to the PM, and VTT and IFE does not yet have an agreement for 2017. However, this is taken into account in the proposal by making this task independent from other tasks. Thus, the possible worst influence will be omission of this task, without any effect on the budget except for the Halden funding.

International co-operation requires flexibility in time schedules, as e.g. test material delivery from the international projects to THELMA may be subject to changes. However, the THELMA project can adjust to this, by putting more focus on other tasks during such waiting periods.

Resource availability is a risk, which cannot be easily predicted at VTT. At VTT, the researchers working in THELMA are also involved in customer assignments, the volume and timing of which is not known far ahead of time and in CNS laboratory refurbishment. The amount of redundancy is rather small at VTT, and is decreasing, which poses a risk for THELMA if key persons leave VTT.

Resource availability is not considered a risk at Aalto University.
References


# THELMA

## TOTAL BUDGET

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume Person months</th>
<th>Expenses Euro</th>
<th>Financing %</th>
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<td>Aalto</td>
<td>Fortum</td>
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<td><strong>WP1 - Ageing of nuclear materials</strong></td>
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Expenses:
- Personnel
- Mat&supp
- Travel
- Ext serv
- Other

Financing:
- VYR
- Aalto
- Fortum
- TVO
- Vattenfall
- Halden
- VTT
- Check

Date: 31.01.2017

Author: Ulla Ehrnstén
### VTT BUDGET

<table>
<thead>
<tr>
<th>WP1 - Ageing of nuclear materials</th>
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<td>T1.2 Ageing of cast SS</td>
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<td>T1.4 PI examination of SS</td>
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</thead>
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<td>T2.2 Alloy 600 RR</td>
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<th>Financing</th>
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</thead>
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<td>3</td>
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<th>Expenses</th>
<th>Financing</th>
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</thead>
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In-kind contributions for of thermally aged material and project results from MIT (USA), KAERI (Korea), Vattenfall (Sweden)

Monetary value difficult to estimate, but e.g. 70.000 hours plant aged material is very valuable

Halden work according to agreement, 420kNok, which is subject to currency variation

NKS application submitted with 250kDr for LOST (80%) and THELMA (20% = 336000Dkr = 6720€)

| VYR | VTT | Aalto | VTT budget, k€ | VYR | Aalto | Fortum | TVO | Vattenfall | Halden | VTT | Tot |
|-----|-----|-------|----------------|-----|------|--------|-----|-----------|--------|-----|-----|-----|
| Anottu | 141 | 103 | 38 | 80 | 0 | 0 | 0 | 7 | 46 | 58 | 191 | 234 | 82 % |
| Myyönnetty | 110 | 80 | 30 | 42% | 0% | 30% | 100% |
| Aalto budget, k€ | 43 | 13 | 0 | 0 | 0 | 70% | 30% | 0% | 0% | 100% |
| 27% | 110 |
SAFIR2018 Project plan

CALL 2017

2.2.2017

Rev. 2

WANDA

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

Project manager: Tuomas Koskinen
Tarja Jäppinen
Miguel Ferreira
VTT Technical Research Centre of Finland Ltd
Fahim Al-Neshawy
Iikka Virkkunen
Aalto University
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1. Research theme and motivation

Since active components of nuclear power plants (NPP) are routinely maintained and repaired, or placed through maintenance programs, the new challenges for long-term operation (LTO) are associated with the deterioration of passive components. Examples of passive component deterioration includes stress corrosion cracking of metal components, radiation induced embrittlement of the reactor pressure vessel, degradation of safety related reinforced concrete structures (containment, spent fuel pools, water channels, etc.), degradation of buried piping and of cables. There is a clear need to integrate non-destructive evaluation (NDE) measurement technologies into proactive ageing management programs (AMP). AMP addresses these issues by focusing on three important aspects of proactive ageing management: early detection of deterioration, monitoring of deterioration, and application of prognostics for the estimation of remaining service life. NDE is one of the recommended tools for the early detection of deterioration of NPP materials.

Safety of a NPP is always in a top priority and the development of the NDE techniques towards more reliable and efficient in-service inspection (ISI) promotes the safety of NPPs. The WANDA project, addressing “NDE on NPP primary circuit components and concrete infrastructure” of the SAFIR2018 programme, is focusing on the development and understanding of NDE methods at different levels.

In Work Package 1 (WP1), research is conducted on the NDE of NPP primary circuit component materials, building on extensive existing experience. WP1 will concentrate on the ultrasonic signal propagation of austenitic stainless steel welds, and the ultrasonic simulation and probability of detection (POD) for a component in the primary circuit of NPP.

In Work Package 2 (WP2), research is conducted on the NDE of NPP concrete infrastructure. The main focus is to bring concrete NDE research on a par with metal NDE. WP2 consists of the evaluation and calibration of the available NDT methods and monitoring systems for NPP concrete structures, by designing and building a representative full-size reinforced concrete wall mock-up. This provides an opportunity for future NDE technique development and a sound basis for the NDE education of engineers focusing on the specific challenges NPP reinforced concrete structures represent.

The work in both WPs will provide support for the ageing management programs of existing NPPs, while simultaneously providing the necessary scientific basis for the design and assessment of new builds.

In the 2017 update to the SAFIR2018 Framework plan is stated that the research on concrete structures is considered important, especially for the LTO of plants. In the framework update studies on aging mechanisms and NDE methodologies from structural design and in-service monitoring point of views for the assessment of integrity and performance of concrete structures are also raised.

This project will introduce new research topics with affinity, the importance of which has been emphasised by the end users. This project also supports the ongoing combined Fortum and TVO monitoring programme of concrete in underground repository conditions, and the proposal for continuation of the BetKYT project in KYT2018 programme.

Both WPs are strongly linked by the common factor – NDE based research, where in fact many methods and technologies are similar but differ slightly according to the application. For this reason, the sharing of knowledge and competence is a vital importance to extend the boundaries of NPP NDE.

A final motivating factor behind the WANDA project is to maintain the high level of expertise of Finnish NDE research, especially concerning NPP component materials, and to raise that of NDE research of concrete infrastructure.

1.1 Background and state-of-the-art

The licensed operating lifetime for existing Finnish NPPs is 40 years, with the possibility of renewal. The initial 40-year design life has been based on economic and antitrust considerations – not on limitations of nuclear technology. As a result, it is possible that some structures, systems and components of NPP have only been designed on the basis of a 40-year service life. The LTO of a NPP requires the periodic renewal of the operating licence, which
in turn requires that NPP demonstrate that their AMP are effectively in control of the ageing of the facilities. This is where ISI programs play an important role in demonstrating sufficient structural integrity of materials and components and guarantee the structural safety, to ensure continued NPP operation in a reliable and safe manner. ISI has an important role in identifying adverse environmental loadings or ageing factor effects before they potentially deteriorate structures compromising the safety of NPP.

Background for the WP1

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 NPP.

Artificial defects are typically used as a reference when the performance of an NDT procedure is demonstrated. The NPP safety regulator guides (YVL) emphasise that the description of defect indications exceeding the recording level shall be given in the inspection report. This information includes definition of defect size, character, location and orientation of defect indications according to ASME Code, Section XI “Rules for in-service inspection of nuclear power plant components” and their comparison with the acceptance level.

According to the previously performed studies in SAFIR2014 on the artificial defects (Leskelä, Koskinen, 2014) the ultrasonic response varies with the type of the defect and also 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.

Austenitic stainless steel (SS) welds are known to be transversely isotropic (Figure 1). This inhomogeneous anisotropic columnar grain structure affects the propagation of the ultrasound and makes its velocity direction-dependent. Scattering of the ultrasonic waves occurs at grain boundaries, which generates both structural noise and attenuation. Like sound velocity in the weld, also the attenuation is direction dependent in an anisotropic medium. This causes false calls and limits the size of the minimum detectable defect due to low signal-to-noise ratio (SNR). Mode conversions and deviation of the ultrasonic wave on the fusion line and columnar grains leads to inaccurate defect sizing and incorrect characterisation and locating of indications (Chassignole et al., 2010; Ploix et al., 2006).

![Figure 1. Transverse isotropic symmetry of austenitic weld (Kolkkoi, 2014).](image)

The known facts e.g. according to NUREG/CR-7019 (Cumblidge et al. 2010) for NDE of NPP ISI are for example:

- Signal-to-noise -ratio between defects of the same size can vary a lot depending on the defect type.
- The crack shape and morphology affects the detectability of the crack when it is skewed.
- Anisotropic austenitic material affects to the propagation of ultrasonic waves and the high noise level reduces the defect detection and sizing capability
- Limited accessibility and complex geometries limit the inspection coverage
- Ultrasonic examination (UT) data analysis requires experience and knowledge of various kinds of indications
- UT is sensitive to planar defects like cracks and lack of fusion
- Mechanised phased array (PA) UT reduces the operating time in radiant environment and facilitates the collection of large amounts of data
- Radiographic examination (RT) has a limited capability of detecting planar defects and has been widely replaced with UT.
Like SS welds, also dissimilar metal welds (DMWs) are challenging for UT because of complex geometries, boundaries, large grain size and anisotropic weld material together with tight and branching service-induced cracks (Cumblidge et al. 2010). DMWs containing Ni-based alloys are susceptible for primary water stress corrosion cracking (PWSCC) and interdendritic stress corrosion cracking (IDSCC) thus the reliable in-service inspections of DMWs is important to ensure the safe operation of NPPs.

NDE simulations are nowadays widely studied and used as a tool to improve the experimental NDE in NPP. Simulation software can take into account various material properties affecting the propagation of ultrasound. The utilization of simulation in studying the inspection problems mentioned above is beneficial. There are four main steps in NDE where simulation methods can be used:

1. Studying the propagation of ultrasound in complex materials and geometries.
2. Early stage development of NDE method. Simulations can be used to study the feasibility of NDE method and to optimize the chosen method for postulated defects.
3. Qualification of NDE method. With simulations, e.g. probability of detection (POD) curves can be produced very cost-effectively.
4. On-site “expertise”. Simulations can be used for the interpretation of complex inspection data and to verify the results.

The performance and reliability of NDE in nuclear industry is paramount. Significant resources are used to confirm sufficient performance through extensive qualification procedures required nowadays around the world. The used qualification procedures have been very successful in confirming highly reliable NDE. However, 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 (RI-ISI). Obtaining reliable POD curves for the nuclear industry has proven difficult and various approaches are explored around the world. In contrast, in the aerospace industry, obtaining POD curves experimentally has a longer tradition and the process was recently (2012) standardized as the ASTM-2862 was published. This standard should be evaluated for feasibility of experimental procedure for the nuclear power plant use. Even if the standard is not directly applicable, it offers the solid basis for developing comparable procedures suited for the nuclear industry.

**Background for the WP2**

An integral parts of NPPs are the concrete-based structures that provide foundation, support, shielding, and containment functions. Reinforced concrete has been used in the construction of NPPs because it is relatively cheap, its structural strength, and its capacity to shield radiation. Important safety related concrete structures of NPPs include the containment building, spent fuel pool, and water channel, among others. Ageing related deterioration of concrete may affect the engineering properties, structural resistance/capacity, failure mode, which in turn may affect the ability of a structure to withstand challenges in service (NUREG/CR-7153). In order to ensure the safe operation of NPPs, it is essential that the effects of potential degradation of the plant structures, as well as systems and components, be assessed and managed during both the current operating license period as well as subsequent license renewal periods. While many mechanical and electrical components can be replaced, this is impractical for many concrete structures. This highlights the need that safety issues related to plant ageing and LTO, with special emphasis on the concrete structures, be resolved through sound scientific and engineering understanding.

Compared to most metallic materials, reinforced concrete is an extremely non-homogeneous material, with a low-density matrix (basically a mixture of cement, sand, aggregate and water), and a high-density reinforcement made up of steel rebar or tendons. NPP concrete structures are often inaccessible and contain large volumes of thick concrete cross sections. These concrete structures are exposed to different environments (moisture, temperature) and varying deterioration mechanisms (chemical reactions, high temperatures, radiation exposure, etc.), which increases the complexity of determining the integrity/quality of concrete.

The LTO of a NPP is not expected to be limited by environmentally related deterioration of the concrete structures, but there have been documented occurrences (Naus 2009), such as: corrosion of steel reinforcement in water intake channels, corrosion of post-tensioning tendon wires, rock anchor/tendon coupling failure, greater-than-anticipated loss of prestressing force due to creep, concrete spalling at containment buttress, water infiltration, cracking and spalling of the containment dome concrete, corrosion of concrete containment liners (interior and areas adjacent to concrete), concrete cracking due to alkali-silica reactions, and leakage of water from spent fuel pools.

While ISI programs have the primary goal of ensuring that the concrete structures have sufficient structural margins to continue to perform in a safe and reliable manner, they must also identify ageing factor effects before they reach sufficient intensity to potentially degrade the structural components. Determination of existing perfor-
formance, the extent and causes of deterioration is achieved through a structural condition assessment, initiating
with a detailed visual assessment of the structure, followed by determination of need for additional surveys or use
of destructive or NDE methods. NPP reinforced concrete structures (RCS) present a unique challenge for deve-
lopment of acceptance criteria because of their large size, limited accessibility in certain locations, the stochastic
nature of past and future loads, as well as that of mechanical and durability performance characteristics due to
ageing and possibly degradation, and the qualitative nature of many non-destructive evaluation methods. Im-
proved guidelines and acceptance criteria to assist in the interpretation of condition assessment results, including
development of probability-based degradation acceptance limits, are required.

The application of NDT methods to NPP RCS has several challenges: wall thicknesses (typically >1.0 m);
dense and complex reinforcement detailing; penetrations or cast-in-place items; limited accessibility (i.e. liners or
other components); severe environments (submerged, radiation); inaccessible structures; limited experience with
NDE methods specifically for NPP; and, lack of specific equipment or knowledge for NDE of NPP RCS.

An updated status of NDT methods and priorities for its development with respect to examination, and instru-
mentation and monitoring of concrete structures in NPP was addressed by the OECD NEA/CSNI WGIAGE work-
shops (WGIAGE 2008, WGIAGE 2013). From these, it was understood that there is a clear need for means of
ensuring concrete structures meet their design criteria, during and immediately following construction. NDE meth-
ods can provide quality control and verification. Furthermore, after the RCS being subjected to ageing degrada-
tion, NDE methods can be used to characterize material properties and ensure adequate performance. There is
still a clear need for NDE methodologies to continue to evolve. For example, international standard specimens
should be developed to allow direct comparisons between various techniques, with consideration given to ensur-
ing a broad range of defects to ensure the Probability of Detection (POD) for a method can be properly deter-
mined. It was also suggested that a round-robin study could be valuable in comparing between NDT methods,
and in determining the variability in the application of the same methods.

In order to address these questions concerning the long term performance of NPP RCS, and to prepare the
needs for future testing and calibration of NDE methods, a multidisciplinary project is proposed addressing sever-
al of the aspects referred to previously. The use of NDE methods is demanding, therefore the need also for an
educational component in order to train and familiarize young professionals with these challenges.

This project will address this problem by providing the means for the long term assessment of NDE and moni-
toring techniques. This is achieved by building a mock-up of a full-size reinforced concrete NPP cross section
(i.e., containment) with the necessary characteristics and details to fulfil the needs of calibration and correlation
of NDE, education of engineers in their use, as well as the development of guidelines for use of NDE techniques.

1.2 Objectives and expected results

The continuous development of NDE methods for ISI is needed. In the WANDA project, this development will
continue addressing the expressed needs of both existing NPPs, and new builds. Main focus of the WANDA pro-
ject is to maintain the expertise level of Finnish NDE research of the NPP component materials and to raise that
of NDE of concrete infrastructure. Additional objectives are:

i. to analyse the ultrasonic wave propagation in austenitic stainless steel welds and further verify the reliability
   of NDE simulations;
ii. to create an experimental POD in particular for nuclear application, e.g. for components in the primary cir-
   cuit,
iii. to participate the international cooperation within U.S. Nuclear Regulatory Commission (NRC) Program to
   Assess the Reliability of Emerging Nondestructive Techniques (PARENT) and its follow-on program;
iv. to critically assess the NDT techniques and monitoring systems currently in use to fulfil the needs of NPP in-
   frastructure evaluation in Finland,
v. to develop guidelines for the use of NDE techniques in design and condition assessment, for the implemen-
   tation of monitoring systems, and for performance based design and ageing management of the concrete in-
   frastructure.

Further the goals of the project are to assess existing and new technologies such as NDE techniques for con-
crete examination and improve the power plants ISI techniques. Results for this four-year project will further
develop NDE techniques used in NPP environment where access and time for inspections is limited. The differences
of the different defects used in the qualification and reference specimens will be evaluated and the
importance of the right defect type and the testing method will be demonstrated. As a result of the international
cooperation within PARENT and its follow-on program, a lot of data, experience and knowledge related to NDE of
DMWs is gathered.
Furthermore, a mock-up of a full-scale NPP concrete cross section with artificial defects will be designed and built in the project. The purpose of the mock-up concrete wall is to allow for continuous long term testing and monitoring (greater than 20 years) which allows for different equipment to be assessed in a well-documented situation.

1.3 Exploitation of the results

The results of WANDA project can be exploited by all the domestic NPP’s both operating and new builds and the information can be shared with foreign partners. NDE is one of the essential areas of research on the safety aspects of the NPP’s.

The relevance of the objectives for NPP, nuclear regulators, and the construction industry in general, is intrinsically linked to the perception of performance though design and modelling on one hand and the condition assessment and ageing of concrete on the other.

As a result, it is expected that:

- Changes will be suggested to the current design procedure to take into account a performance based approach to design and assessment of concrete. This requires close cooperation with the NPP, contractors, consultants, material producers, regulators, etc.
- Ultimately NPP RCS will benefit from increased service life. This alone is quite significant because the repercussions are many, and directly linked to the sustainability of the sector: reduction in the consumption of natural resources; reduction in the production of construction and demolition waste; reduction in the production of CO2 as a result of the previously mentioned factors
- The information of the material properties and propagation of ultrasound is important e.g. for qualification processes and realistic simulation of UT.
- Realistic simulation of UT is an efficient tool to aid the NDE design and qualification processes.
- Participation to the international programs and their results will support the development of the best NDE practices
  - NUGENIA consortium (e.g. NUGENIA+ project MAPAID)
  - PARENT Atlas information tool for the ISI of DMWs.
  - ODOBA Research project on the durability of RCS (IRSN and international consortium);
- Gained knowledge will be used also on Euratom Horizon2020 call project preparation background.
- Educate new engineers (expert in NDE) for the Finnish NPPs and industry.

The research results will also be published in peer-reviewed journals and refereed conference proceedings. The research outcomes will be broadly disseminated to the end users, decision makers and other stakeholders, such as nuclear regulators, the utilities as well as the scientific community and general public via seminars.

1.4 Appropriateness of the project to SAFIR2018 programme

The safety of the NPPs is improved by developing more efficient and reliable NDE techniques. The SAFIR2018 framework plan also emphasises the importance of the NDE research on the field of the nuclear energy. Also the importance of developing the NDE methods was highlighted in the assessment of the SAFIR2014 research programme.

SAFIR2018 – Update to the Framework plan for 2017 call is stating that in SG3 the research on concrete structures is considered important, especially for the LTO of plants, e.g. studying aging mechanisms and NDE methodologies from structural design and in-service monitoring point of views for the assessment of integrity and performance of concrete structures. It is stated also on SG3 that research on NDT generally was considered important.

This project proposal supports SAFIR 2018 goals directly in several topics in the structural safety and materials research area, e.g. the non-destructive examination and assessment methods (3.4.4.3) and the evaluation of the performance of the NPP safety-related concrete structures (3.4.4.4).

In the framework plan it is stated in section 3.4.4.3 that the important research areas for metallic structures include the study of the inspectability of fatigue fractures with NDT methods and, in particular, the verification of the reliability and detection probability of the related observations and the creation of probability of detection (POD) graphs. A topic where NUGENIA collaboration has been active and most likely will continue is the ultrasonic wave propagation in stainless steel and dissimilar metal welds. In addition, POD modelling should be developed for complex objects, such as reactor pressure vessel (RPV) assembly and DMWs.
The project also aims to comprehensively cover the design, construction and ageing assessment of mock-up of a full-size reinforced concrete wall require expertise from several fields of engineering. The project can be considered to be inherently multidisciplinary and integrating.

1.4.1 Cooperation to national and international projects

The research in the WANDA project has links to other SAFIR2018 projects. DMWs are a topic also in LOST (LTO aspects of structural integrity) and the aspect of POD is brought up also in FOUND (Analysis of fatigue and other cumulative ageing to extend lifetime). WANDA has the NDE aspect in all of the topics and therefore the research will not be the same despite of the same topics.

PARENT program established by the U.S. NRC has focused on the research of the reliability of NDE of DMWs. The goal of the program is to investigate the effectiveness of current and novel NDE procedures and techniques to find and evaluate defects in nickel-alloy welds and base materials. The Pacific Northwest National Laboratory (PNNL) will develop PARENT Atlas information tool which will be delivered to PARENT participants. The Atlas will include information on the PARENT program together with international experience from PARENT participants.

Other international project that VTT has participated and the results are followed by WANDA is the NUGENIA+ project MAPAID (Modelling and Application of Phased Array ultrasonic Inspection of Dissimilar metal welds). MAPAID aims to model and validate phased array ultrasonic techniques (PAUT) for NDE of DMWs of NPPs. This project enables to quantitatively assess the contribution of phased array techniques to improved NDE performances of such parts, as well as the ability of simulation to help for design, optimization and interpretation of such inspections.

For Euratom Horizon2020 call 2016, close cooperation with NOMAD (Nondestructive Evaluation System for the Inspection of Operation-Induced Material Degradation in Nuclear Power Plants) project consortium have been made to build a project plan for NDE – tool for reactor pressure vessel.

The work in the WP2 has been divided into the current format because some preliminary groundwork is required before the detailed technical difficulties of design, construction, monitoring and testing can be addressed, solved accordingly. In this respect the project is planned to interact with similar projects such as the Finnish Public Nuclear Waste Management Program – KYT2018; the Swedish Barsebäck NPP – Decommissioning project; IRSN’s ODOBA - Research project on the durability of RCS (including an international consortium: US NRC, Canadian CNSC, Belgium Bel-V, Chinese NCS, and many French universities), EPRI – Electric Power Research Institute (in collaboration with their own concrete NDE program), ORNL - Oak Ridge National Laboratories (experienced with NDE and SC construction typology), and BAM Civil Engineering Departments (world leading in concrete NDE). The project looks to interact with other research institutes and universities, and actively participate in the technical committees of international concrete research groups.

The project also aims to develop a new and ambitious educational component, and exploit a cooperation network within the IAEA in order to draw on international expertise in the field of concrete performance and ageing management. It is a jointly developed effort by VTT Technical Research Centre of Finland Ltd. and Aalto University. This fulfils the cooperation between research institutes and universities.

1.5 Education of experts

One important factor for the future is the transfer of know-how in the area of NDE to a younger generation of scientists. WANDA -project for SAFIR2018 is one of the important channels to transfer the knowledge to the younger generation and thus support the education of new high-level experts for nuclear area and to link these young scientists to international co-operation.

Also new areas of expertise are raised to the Finnish NDE field as a result of the WANDA-project. These topics include e.g. NDE simulations, NDE-reliability calculations and basic NDE of concrete structures.

During the first two years, topics of a master’s thesis work have raised in the area of artificial defect NDT and simulation.

In the WANDA project two post-graduate research scientists will work also on their PhD thesis.

The joint research between VTT and Aalto University enables the close work with under graduate students to introduce them to the field of nuclear energy as master’s thesis work or as researchers.

During the third year, two under graduate students will carry out the design, implementation and NDT detection of simulated defects of thick-walled concrete structure.
2. Work plan

The project of non-destructive examination of NPP primary circuit components and concrete infrastructure has two work packages. The WP1 is about NDE of the primary circuit components, the welds and detection probability studies and WP2 is concrete infrastructure NDE.

![Diagram of WANDA](image)

Figure 2. The Structure of the WANDA.

The topics in WANDA will be presented in different international conferences, seminars and workshops related to NDT during the year 2017. Presentations will be related to research made in WANDA in 2017 or earlier.

2.1 WP1 - NDE on NPP primary circuit components


The WP1 subtasks during the years 2015 – 2018 are concentrating on evaluating the propagation of ultrasound in the welds of primary circuit components, simulation of NDE and assessing NDE reliability in NPP.

Simulation is valuable and cost-effective tool when developing and improving the NDE procedures to detect defects in NPP primary circuit components. To make realistic simulations, the material properties and the propagation of the ultrasound in the material needs to be studied which is rarely done. Data of microstructure, elastic constants, velocity and attenuation is therefore needed. This information can be obtained with a weld sample using ultrasound (e.g. full matrix capture (FMC), sound velocity and attenuation measurements) together with macrographs and electron backscatter diffraction (EBSD) technique. These efforts would provide valuable information on modelling the austenitic weld into the simulation software.

Simulations will be verified based on the data of phased array ultrasonic techniques (PAUT) examinations made during 2014–16 in WANDA project. With NDE simulations, the range and amount of defects can be easily increased to the necessity of developing the inspection techniques and creation of POD curves.

There is a possibility that POD curves created in WP1 would be used in the break probability calculations made in SAFIR FOUND project. The realistic POD curves are valuable tool for risk-informed ISI calculations.
VTT has participated to U.S. NRC PINC (Program for the Inspection of Nickel Alloy Components) and PARENT programs on the NDE of DMWs. The ongoing PARENT program comprises a study of the efficiency of different commercial and emerging NDE techniques to detect and size SCC defects in Ni-base alloy DMWs. Participating in the PARENT and its follow-on program gives a privilege to get the latest data and the knowledge of current and emerging NDE techniques to support the development of best NDE practices for the ISI of DMWs. WANDA follows closely this international project. It is noted also on the SAFIR2018 framework plan that U.S. NRC PARENT should be followed.

**WP 1 plans for the year 2017**

Table 1. Person months allocated in WP1.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>Partner 1 VTT</td>
<td>1.5</td>
</tr>
<tr>
<td>Partner 2 Aalto University Department of Engineering Design and Production.</td>
<td>1.5</td>
</tr>
</tbody>
</table>

2.1.1 Task 1 Propagation of ultrasound in austenitic SS and DMW welds (T1.1)

The propagation and attenuation of the ultrasound in austenitic stainless steel and its weld is not straightforward. Welds tend to vary from each other and therefore also the microstructural properties vary from weld to weld. Though, some tendency can be predicted and with more testing and intent research on different materials this question can be opened. This task aims at studying the propagation of ultrasound in austenitic stainless steel pipe weld and to make realistic simulations of ultrasonic testing of the weld. In the year 2017 the characterisation and modelling of austenitic stainless steel weld is started. The finalisation of this task is conditional to the project continuation on year 2018.

By the end of the year 2018 the following topics are studied.
- Characterisation
  - The weld microstructure and acoustic properties are characterised.
- Simulation
  - Modelling the weld and ultrasonic signal propagation with simulation software
  - Simulation of inspection

The characterization of the austenitic steel weld requires among others macrographs EBSD, elastic modulus, density, attenuation and sound velocity studies to specify the simulation parameters. The sample for weld destructive and non-destructive characterisation will be taken out from the solid part of the weld of the test pipe prepared in WANDA 2015. The results are used for the more precise modelling of the austenitic steel weld and the ultrasonic inspection of it.

2.1.2 Task 2 Assessing NDE reliability – experimental and model assisted POD for the nuclear industry (T1.2)

While the existing NDE qualification has been very successful on the whole, the gained experience has also highlighted significant shortcomings of the current approach. Firstly, recent shift towards more optimized risk-informed inspections have made it increasingly important to have reliable and quantitative assessment of NDE reliability. The risk-reduction attributed to NDE is highly dependent on the expected probability of detection (POD) achievable. The POD depends on the used method, personnel and other factors. However, quantitative and reliable evaluation of POD has remained challenging for the nuclear industry (despite being commonplace and standardized in the aerospace industry). Secondly, with the history of qualified inspections now extending over 20 years and thousands of inspections around the world, it’s probable or even expected that some misses occur even with high-quality and qualified inspection procedures. Indeed, recently for example EPRI (Spanner, 2015) have reported leaks resulting from cracks missed in previous qualified inspection. The deterministic nature of the
current qualification scheme makes it difficult to assess these cases. Having quantitative, probabilistic information regarding the NDE methods would greatly help re-assessing NDE performance based on in-service experience and avoid the potential and costly recall of related NDE qualifications.

The long-term aim for this task is to evaluate and develop methods for quantitative experimental determination of NDE reliability in general and POD in particular for nuclear applications. The work is based on current nuclear qualification procedures and existing aerospace standards (e.g. ASTM E2862).

The first steps on this new subject taken during 2016 with cooperation between Aalto University and VTT was to evaluate the current methodologies for experimental POD determination in view of their application in the nuclear industry and have preliminary POD with UT-data file modification and simplified online tool. The results showed that the data file modification shows great potential for acquiring representative POD-curves for nuclear industry applications. Furthermore, the approach enables graded approach to POD estimation spanning from simplified online-tools for training and first semi-automated estimation of POD performance all the way to full POD evaluation using modified data files analysed with the proprietary analysis software used during the actual analysis.

While the results in 2016 are very promising, they were obtained with very limited set of actual cracks and data files and thus need to be verified with more data. For 2017, the task is to produce test blocks with thermal fatigue cracks of sufficient variety to span the interesting range for POD determination. With data from the cracks, the results of 2016 can be confirmed with actual un-modified crack indications. Furthermore, the simplified online tool is augmented with additional views to make it more representative to the actual data analysis process.

As in 2016, in-kind contribution from Trueflaw is expected, including crack manufacturing and use of software necessary for UT-data file modification and use of online tools for POD determination.

In 2017 the milestones for T1.2 are:

- Data acquisition of additional data files for data modification
- POD analysis on the data gathered from online tools and modified data analysis (Aalto)

The acquired insight into nuclear industry POD curves and the resulting POD curves will be published in scientific paper (Deliverable D1.2.1). Due to budget cuts the work will be completed with reduced volume.

2.2 WP2 – Containment wall testing for long-term durability


The research project has four sub-projects that are developed during the course of the quadrennial 2015 – 2018. These are:

- Subproject 1. Non-destructive testing and evaluation (NDT&E) methods and monitoring of concrete performance
- Subproject 2. Design and construction of the mock-up of a full-size reinforced concrete wall
- Subproject 3. Evaluation and calibration of NDT&E methods
- Subproject 4. Evaluation and calibration of monitoring methods
Figure 3. Schematic representation of the structure of WP2 and the interaction between the subprojects

Subproject 2.1. NDT&E methods and monitoring of concrete performance

Produce a state-of-the-art summary of the understanding of NDT&E methods for RCS in Finland. This work can draw on previous studies. The work will focus on the advantages and disadvantages of different methods, and include guidelines for the selection of methods and understanding the unique and complementary characteristic of measurements/methods. Guidelines shall also address interpretation and use of results for design and assessment purposes.

Produce a second state-of-the-art summary covering research projects that have addressed NDT&E inspections of NPP concrete structures, with focus on heavily reinforced, thick-sections, reduced access, mock-up design options (geometry, reinforcement disposition, defects, conditioning, induced deterioration, etc.).

Produce a third state-of-the-art summary addressing the monitoring of RCS performance, with focus on the sensors and technologies currently available. Special focus lies on reliability, durability and longevity of such sensors and monitoring systems. The work will include an assessment of the technology readiness level\(^1\) of these items, with regard to their use in nuclear structures. Since an effective data management is essential for ensuring monitoring quality, different ways of data management, including all steps from collection to processing and storage of data are being reviewed.

These three state-of-the-art reports will provide the basis for defining the design criteria for the mock-up of a full-size reinforced concrete wall (geometry, concrete characteristics, reinforcement, testing, monitoring, type and layout of sensors, location and size of defects, etc.)

**Education dimension:** Coursework will be developed by Aalto University and VTT to provide the basis for the planning of an introductory course with focus on engineering aspects of NDT&E, with special focus on NPP RCS.

**Cooperation dimension:** Participation in the NEA/CSNI WIAGE group and extending cooperation with other interested parties in Sweden (SSM and Vattenfall) and across Europe.

Subproject 2.2. Design and construction of the mock-up of a full-size reinforced concrete wall

This subproject deals with the design and construction of the mock-up of a full-scale concrete wall. Consideration on the criteria for design include the geometry, reinforcement types and displacement, concrete material characteristics, inclusion of defects, location, accessibility and exposure conditions, etc.

Furthermore, the types on NDT&E methods, sensors and monitoring system are also taken into account in the design. This task builds heavily on the previous subproject, but is also linked to subproject 5 which addresses all aspects of performance based design and assessment RCS.

\(^1\) See http://www.nasa.gov/content/technology-readiness-level/
Cooperation dimension - Cooperation might include short missions to visit test rigs developed at ÚJV Řež and LPI, and discuss details of their testing program. Participating in the NEA/CSNI WIAGE group and extending cooperation with other interested parties in Europe.

Subproject 2.3. Evaluation and calibration of NDT&E methods
This subproject addresses several aspects related to the NDT&E methods for condition assessment of RCS. A testing program will be set up which will cover calibration of test methods, correlation between test methods, effect of time dependency and testing conditions on test methods, accuracy of test methods, among other aspects yet to be defined. The focus will be on the available and new test methods. The purpose of the mock-up of a full-scale concrete wall is to allow for continuous long term testing (greater than 20 years) which allows for different equipment to be assess in a well-documented situation.

Furthermore, the mock-up of a full-scale concrete wall will allow owners of NDT&E equipment to test and calibrate performance accordingly.

Cooperation dimension - Cooperation might include participation of national and international consultant companies proving NDT&E services, and the providers of NDT&E equipment to show case their equipment and developments, both during research and when the technology become commercial.

Education dimension - The mock-up of a full-scale concrete wall will allow for the training of young engineers in the use different NDT&E equipment.

Subproject 2.4. Evaluation and calibration of monitoring methods
This subproject addresses several aspects related to the monitoring methods for performance assessment of RCS. A testing program will be set up which covers calibration and validation of test methods, correlation between different test methods, effect of time dependency and testing conditions on test results, resolution and accuracy of test methods and repeatability of measuring results, among other aspects yet to be defined. The focus will be on the available and new test methods and techniques. The purpose of the mock-up of a full-scale concrete wall is to permit continuous long term monitoring (greater than 20 years) which allows for different equipment to be assessed in a well-documented situation (reliability, long term stability, etc.).

Cooperation dimension - Cooperation might include participation of national and international consultant companies proving monitoring sensors and equipment, and the providers of monitoring services to show case their equipment and developments, both during research and when the technology become commercial.

WP2 plans for the year 2017

Table 2. Partners and person months allocated to WP2 in 2017.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Partner 1 – VTT</td>
<td>3</td>
</tr>
<tr>
<td>Partner 2 – Aalto University Department of Civil and Structural Engineering</td>
<td>4.2</td>
</tr>
</tbody>
</table>

From Figure 3, where an overview of WP2 for the 2015-2018 period is given, can be seen that the focus for the year 2017 is in the design and construction of the mock-up of a representative reinforced concrete element from a NPP. The work has been divided into two phases: a detailed design phase where details of the mock-up is finalised, including the NDT tests defects and the monitoring system to be used; and a final phase where the mock-up element is built.

Furthermore, international co-operation with similar research projects that are addressing NDT&E inspections and monitoring systems of NPP RCS will continue to be actively pursued.

2.2.1 Task 1 - Preparation of the construction of the thick-walled reinforced concrete mock-up (T2.1)

This task is divided into two parts: a) design, implementation and detection of simulated defects of TVO thick-walled concrete structure shown in Figure 4 and b) preliminary tests for the construction of the thick walled concrete mock-up. In the first part, realistic defects have designed in Aalto University and are implemented in the TVO thick concrete wall, which simulate the common defects during the construction process and cumulative degradation of the concrete with time. The representative defects of construction process are i) honeycombing, ii)
air filled voids and iii) water filled voids. The representative cumulative ageing defects of the wall are i) cracks and delamination of concrete and ii) corrosion of the reinforcement steel. The proposed embedded defects in the TVO-thick concrete wall are designed to be identifiable by a majority of the current state-of-the-practice NDT methods. The work in this part of the task will serve as background for the construction of the WANDA thick-walled concrete structure.

Figure 4. Schematic overview of the TVO - thick-walled concrete structure.

Based on first part results and the reports from task 2.1 and task 2.2, the second part of this task will overlook the construction of the mock-up. The concrete wall will simulate the following NPP concrete structures: (i) reinforced concrete structures – RCS, (ii) pre-stressed concrete structures – PCS and (iii) steel-plate composite structures - SPCS. The concrete wall is stored in indoor condition protected from wind and rain and the other part is stored in the outdoor condition. Typical defects existing in real reinforced and pre-stressed concrete structures as, e.g. honeycombs, delamination and cracks, and corrosion of steel members, will be simulated in the wall for later calibration of the NDE testing equipment. A reduced system for long term monitoring devices (sensors) will be installed in the concrete wall for monitoring the ageing and performance of the structures, e.g. the relative humidity, temperature, corrosion rate, and concrete strain.

2.2.2 Task 2 – Probability of Detection methodology applied to concrete NDE (T2.2)

For the in-service condition assessment of reinforced concrete structures, it is necessary to qualify and quantify the extent of the defects using a range of NDT tools and methods. The analysis and reliability check of the data produced by these tools and methods gives a large amount of information over a long time period. For evaluating the capability of these NDT tools and methods, the determination of the Probability of Detection (POD) is an essential methodology (Kessler, 2015). The capability is defined as the probability of detecting a defect with a particular size under specified inspection conditions and a defined procedure.

The work of this task includes:
- An overview of the current state-of-the-art in using the probability of detection in the field of reinforced concrete structures condition assessment. (D2.2.1)

2.2.3 Task 3 – Detailed design and construction of non-destructive evaluation and monitoring mock-up (T2.3)

Research needs have highlighted the importance of specially designed reinforced concrete mock-ups, especially replicating realistic 1-1 size components of NPP reinforced concrete structures that can provide realistic flaws that are similar to actual flaws in terms of how they interact with a particular NDE technique. Reinforced concrete mock-ups allow the isolation of certain detection problems (flaws, defects, deterioration) as well as the variation of certain parameters. Because of the controlled construction conditions, the number of unknown variables can be decreased, allowing specific focus to gain further information on the capabilities and limitations of the NDE methods. To minimize the significant influence caused by boundary effects, the dimensions of the mock-up
should be representative of large heavily reinforced cross sections, so as to allow for comparative testing and to evaluate the state-of-the-art NDE techniques in this area and to identify additional developments necessary to address the challenges potentially found in NPPs.

As mentioned previously, such unique mock-ups will be useful for calibration and validation of new technology and processing techniques, in addition to the education of young professionals and possible accreditation of NDE specific engineers.

The detailed design in this task will focus on the following conceptual design criteria defined in the outcome of last year’s WANDA WP2 deliverable. Specific focus will be given to:

- boundary effect concerns (proximity of defects – affects wall size/dimensions)
- large specimen concerns (problems large walls can create – construction, personal safety)
- proposed specimen detailed design (dimensions, reinforcement, liner (?), formwork stability, concrete composition, simulated defects (types and location), monitoring system (sensors, location, data transfer & storage), and other construction details (e.g.: possibility of tendons/ducts for grouting, etc.)

As mentioned previously, typical defects existing in real reinforced and pre-stressed concrete structures as, e.g. honeycombs, delamination and cracks, and corrosion of steel members, will be simulated in the wall for later calibration of the NDE testing equipment. A reduced system for long term monitoring devices (sensors) will be installed in the concrete wall for monitoring the ageing and performance of the structures, e.g. the relative humidity, temperature, corrosion rate, and concrete strain.

During this task, the initial state of the wall (time 0) needs to be known as precisely as possible. Therefore, the reinforcement and all installations (sensors, wires, defects) will be documented by e.g. laser scanning, radiographic testing. A laboratory testing program will be set up to cover the time-dependent material characteristic properties of the fresh and hardened concrete.
3. Deliverables 2017

The planned deliverables for 2017 are listed in the table 3.

Table 3. List of deliverables

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.2.1</td>
<td>Scientific paper of Nuclear industry POD curves on ultrasonic testing.</td>
<td>1.5</td>
<td>15.12.2017</td>
</tr>
<tr>
<td>D1.2.2</td>
<td>Conference presentation on reliability of NDE</td>
<td>0.1</td>
<td>8.9.2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Report on a) the definition of the condition zero for the mock-up wall and b) the detection of the imbedded defects in TVO-thick wall.</td>
<td>1</td>
<td>15.06.2017</td>
</tr>
<tr>
<td>D2.2.1</td>
<td>Report on the current state-of-the-art in using the probability of detection in the field of reinforced concrete structures condition assessment</td>
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<td>15.10.2017</td>
</tr>
<tr>
<td>D2.3.1</td>
<td>Report on design of mock-up: NDE and monitoring</td>
<td>4.8</td>
<td>15.12.2017</td>
</tr>
</tbody>
</table>

| Total pm | 8.7 |
4. Project organisation

The organisation responsible for the WANDA project is VTT. The Project manager of WANDA is Tuomas Koskinen from VTT (BA2507). The work package 1 of WANDA is managed by Tuomas Koskinen and the manager for WP 2 is Miguel Ferreira, VTT (BA2402).

Aalto University is participating on WANDA with two different departments. The responsible contact on WP 2 at Aalto University is D.Sc. (Tech.) Fahim Al-Neshawy. And the responsible contact in WP 1, Task 1.2 at Aalto University is prof. Iikka Virkkunen.

Table 4. Researchers, their organisation, the tasks they are contributing, and the estimated person months in WANDA project in the year 2017

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tuomas Koskinen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>Project manager, T1.1</td>
<td>0.5</td>
</tr>
<tr>
<td>Tarja Jäppinen</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>0.25</td>
</tr>
<tr>
<td>Esa Leskelä</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>0.5</td>
</tr>
<tr>
<td>Jonne Haapalainen</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T1.1</td>
<td>0.25</td>
</tr>
<tr>
<td>Miguel Ferreira</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>T2.2 – T 2.3</td>
<td>2</td>
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<tr>
<td>MM</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.2 - T2.3</td>
<td>1</td>
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<tr>
<td>Fahim Al-Neshawy</td>
<td>Staff Scientist</td>
<td>Aalto University</td>
<td>T2.1 – T 2.3</td>
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<td>MM</td>
<td>Research Trainee</td>
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<tr>
<td>Iikka Virkkunen</td>
<td>Professor (Adjunct)</td>
<td>Aalto University</td>
<td>T1.2, T2.2</td>
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<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>10</strong></td>
</tr>
</tbody>
</table>
5. Risk management

One of the resources for WANDA will be the master’s thesis workers. If the proper thesis workers are not found the work load for current project personnel is too much. Also the training of a new project member is time consuming and without proper planning can take too long.

The application of NDT to NPP reinforced concrete structures has several challenges i.e. the infrastructure wall thicknesses is typically more than one meter thick with dense reinforcement. To find an appropriate NDT method to inspect the containment could be challenging.

VTT will detail the measures to be implemented in order to eliminate or mitigate the risk of his failing to meet programmed requirements. This will include but not be limited to:

- Preliminary risk analysis and assessment report of expected impact on cost, performances and schedule (Risk Card)
- Review of performance against approved programme and actions to identify potential development of delays
- Associated list of actions to implement in order to reduce the risk exposure
- Procedure to maintain the above documents up to date throughout the execution of the Work.

The project’s risk management plan is shown in Table 5.

Table 5. Risk assessment and mitigation plan.

<table>
<thead>
<tr>
<th>Risk</th>
<th>Probability of occurrence</th>
<th>Potential impact on project success</th>
<th>Mitigation Plan</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plan related: Significant changes in the research plan.</td>
<td>Low</td>
<td>Medium</td>
<td>Study the impact of the changes on schedules and results Implement changes, if the impact is high</td>
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<tr>
<td>Equipment failure:</td>
<td>Low</td>
<td>High</td>
<td>Regular equipment inspection, maintenance and management of change.</td>
</tr>
<tr>
<td>Resource related: Costs could rise significantly during the time of the project</td>
<td>Low</td>
<td>High</td>
<td>Using other funds.</td>
</tr>
<tr>
<td>Human and management related: Loss of key researchers (unable to complete key tasks)</td>
<td>Low</td>
<td>High</td>
<td>Identify alternative resources in case of unexpected absence Ensure complete records of work are available at any point</td>
</tr>
</tbody>
</table>
References

ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, Division 1 (ASME Code, Section XI)


Spanner, J., Improving Ultrasonic Examination Procedures for Detection of Thermal Fatigue. Fourth International Conference on Fatigue of Nuclear Reactor Components, 1st October 2015, Seville, Spain

YVL guides E.5 In-service inspection of nuclear facility pressure equipment with non-destructive testing methods, 20 May 2014

Working Group on Integrity and Ageing of Components and Structures (IAGE). Workshop on Ageing Management of Thick Walled Concrete Structures, ISI, Maintenance and Repair, Instrumentation Methods and Safety Assessment in View of LTO. Prague, Czech Republic, 1-3 October 2008

Working Group on Integrity and Ageing of Components and Structures (IAGE) OECD/NEA WGIAGE Workshop on the Non-destructive Evaluation of Thick-walled Concrete Structures Prague, Czech Republic, 17-19 September 2013
Nondestructive testing of nuclear power plant primary circuit components and concrete infrastructure

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
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</thead>
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<td></td>
<td>Volume</td>
<td>Personnel</td>
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<td>welds</td>
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<td>assisted POD for the nuclear industry</td>
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<tr>
<td>T2.2 - Probability of Detection methodology applied to</td>
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<td>concrete NDE</td>
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**Check sum:**
52.9 31.4 21.5 107.2

**Comments:**
Other costs: VTT research area expenses
Mat&supp: T1.2 part of CIVA license, WP2 wall mock-up specimens

**Travel:**
European NDT cooperation, conferences e.g. 7th European-American Workshop on Reliability of NDE, dissemination, networking e.g. Nugenia Forum
<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Volume</td>
<td>Personnel</td>
</tr>
<tr>
<td></td>
<td>person months</td>
<td>keuro</td>
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<td>17,3</td>
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<td>T1.2 Assessing NDE reliability – experimental and model assisted POD for the nuclear industry</td>
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<td>WP2 - NDT testing of concrete</td>
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<tr>
<td>T2.1 - Preparation of the construction of the thick-walled reinforced concrete mock-up</td>
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<td>T1.2 Assessing NDE reliability – experimental and model assisted POD for the nuclear industry</td>
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<td>WP2 - NDT testing of concrete</td>
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INFRA – Research infrastructure:
SAFIR2018 Project plan

INFRAL

Development of thermal-hydraulic infrastructure at LUT

Joonas Telkkä, Heikki Purhonen, Antti Räsänen
Lappeenranta University of Technology
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      2.4.1 Project management and publications (T4.1) .......................................10
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1. Research theme and motivation

The up-to-date experimental research infrastructure is essential for the modern nuclear safety analyses. The implementation of novel measuring techniques in the thermal hydraulic experiments is needed for the validation of the Computational (Multi-)Fluid Dynamics (C(M)FD) methods [Prasser, 2008, Rahman et al., 2009]. During the last decade, the use of CFD methods has become more common in the safety analyses of nuclear power plants. In order to rely on those analyses, one needs to have the credible validation of the method against experimental data from the CFD grade measurements. The reason for the growing popularity of CFD analyses is the complexity of many thermal hydraulic phenomena that cannot be accurately predicted using the one-dimensional system codes [Bestion, 2014, Bestion, 2015]. Since more complex phenomena also requires more advanced experimental facilities, the same CFD tools can be used in the design of those facilities. This interconnection improves the overall performance of the experimental setup and validation. The general acceptance of the CFD codes for the safety analyses requires that the tools have been properly validated and the applied models and the methods accepted [Bajorek, 2015, Boyd, 2015, Blömeling and Schaffrath, 2015]. The validation experiments differ from the typical thermal hydraulic experiments as the level of detail is a one-step higher to properly validate the model(s) in question [Smith, 2015].

In the SAFIR2014 programme, the research project ELAINE was launched for the enhancement of measurement instrumentation available for the nuclear safety experiments in Lappeenranta University of Technology (LUT). Significant milestones in the project were the acquisitions of Particle Image Velocimetry system (PIV), Wire-Mesh Sensor (WMS) electronics and the system of three modern High-Speed Cameras (HSC). In addition to acquisitions of experimental hardware, the new data storage system for the experimental data (EDS) was developed and taken into active use. In addition, one important task in the project was the maintenance of (PWR) PACTEL test facility to secure its operability and availability for the experiments.

In the SAFIR2018 programme, the INFRAL project was launched in 2015 and it aims to 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 PACTEL test facilities and to launch a study on the new major test facility to prepare for the post-PACTEL era. Demonstration of the applicability of measuring methods and instrumentation for a certain application is also important before launching a special research project.

1.1 Background and state-of-the-art

The acquisitions of advanced measurement systems to LUT have been carried out during the last few years in a timely step-wise manner. The measurement techniques were taken to the active experimental use in the same fashion. However, the use of advanced techniques always requires the building up of the expertise and learning so-called “tricks of the trade” i.e. plug-and-play operation and data extraction are rarely possible.

Post-processing of the measurement data is an important part of the research workflow and the development of the data analysis methods is needed if there are no readily available computational tools. Commonly, in-house developed tools are applied to extract the essential information from the WMS and HSC data. In PIV, the post-processing software is typically supplied to the user by the manufacturer of the PIV system. Along the hardware acquisitions, the development of the in-house data analysis tools has been carried out and it will be continued in the proposed project. The data analysis tools can be quite research problem specific, which means that they have to be customized and further developed for different experimental arrangements. For example, the same numerical procedures applied to analyse the void fraction measurement with WMS cannot be directly applied to extract the data from the flow mixing experiments.
Research infrastructure at LUT is being continuously developed to have the state-of-the-art capabilities for Finnish experimental use. Also the advanced applications for the techniques are actively studied and tested to provide new means for measuring quantities of interest. The issues and the challenges faced in the thermal hydraulic safety research and the code development may have partly changed over the past years, but some eternal questions still have remained and are yet to be resolved [Yadigaroglu and Lakehal, 2015, Song, 2015].

1.2 Objectives and expected results

The goal of the INFRA project is to ensure the availability of the infrastructures and research teams capable to design, construct and operate test facilities representing the physics of nuclear safety related phenomena with sufficient accuracy. Adopting and testing new, advanced measurement techniques enables to produce high quality test data for the development and validation of modern computational tools.

Adopting and testing the advanced and combined use of new measurement techniques allows targeted research projects to achieve high quality data with using pre-tested configurations of instrumentation and measurement systems. Ability to apply the combined use of different advanced systems or using WMS in a new application are examples of the project outcomes.

The maintenance work of (PWR) PACTEL consists of the maintenance of the hardware of the facility (piping, vessels, and inspections) as well as the transducers and other instrumentation and the data acquisition system. By this work, the operability of the facilities will be ensured. Besides the (PWR) PACTEL, the laboratory has several control and data acquisition systems, which occasionally require spare parts or even reserve parts to make sure that those parts are available when needed.

1.3 Exploitation of the results

It is foreseen that the results from all the activities performed in the proposed project can be widely applied to experimental research performed in LUT and in Finland. The measurement techniques acquired, tested and developed in the project are available for SAFIR and other projects that conduct tests at LUT laboratories. The (PWR) PACTEL facility is in active use in SAFIR projects, and internationally as well [Rikkonen et al., 2013, 2014]. The INFRA project ensures that advanced instrumentation and access to integral test facilities is possible also in the future.

1.4 Appropriateness of the project to the SAFIR2018 programme

The proposed project is well suited to the SAFIR2018 programme as the scope of the research is on the development of the experimental research infrastructure in LUT to meet Finnish nuclear safety research needs and to keep it up-to-date to meet the modern standards in the measurements. The recommendations made by the international evaluators of the SAFIR2014 programme support the previous statement [MEE, 2014].

1.5 Education of experts

The young researchers and the research trainees working in this project will gain expertise in the state-of-the-art measurement techniques and their applications in the nuclear safety research. The project offers good topics for bachelor's and master's theses such as the development of new applications for the WMS technique and optical methods.
2. Work plan

The work packages of the INFRAL project are aiming to ensure the availability of the resources related to the produce of experimental data using test facilities, up-to-date methods and human resources. The measurement techniques applied in thermal hydraulic experiments are developing continuously. Following the development, acquiring the new methods and developing them yourself, helps to get better quality data from various test configurations. The overall plan of the INFRAL project is presented in Figure 1. The content of the WP1 is revised yearly to respond better to the needs of the research. The change to the prior plan is highlighted with the different colour. The WP2 and WP3 are related to the limited lifetime of the current integral test facilities in the LUT laboratory and to finding a solution to handle the situation.

The INFRAL work plan for 2017 follows the initial volume planning developed in late 2014 for the whole duration of SAFIR2018. Now that the Finnish Academy of Sciences has granted some funding for the development of next generation thermal-hydraulic test loop, INFRAL would be ready to accelerate MOTEL design and implementation activities, starting even some hardware investments already in 2017.

2.1 Advanced measurement techniques (WP1)

The work package includes activities that are related to the use of advanced measurement techniques in LUT. Part of the work is to develop analytical tools to extract the needed data from the measurements. The other part of the work is to study the applicability of the techniques for different flow problems and to develop new measurement solutions.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>9</td>
</tr>
</tbody>
</table>

2.1.1 Advanced and combined use of PIV/WMS/3D Cam systems (T1.1)

The task supports the use of advanced measurement techniques in LUT as new experimental procedures for flow measurements are being developed and tested. In 2016, the main activity with the PIV system has been the use of the PIV hardware with add-on components for the spray experiments to support the INSTAB project by
providing the measurement setup. These so-called shadowgraphy measurements have been performed to record the physical parameters of the laser-backlighted water droplets to provide better boundary conditions for the simulations. A master’s thesis has been written regarding the shadowgraphy measurements. The master’s thesis process is in the fine-tuning phase.

In 2016 the PIV measurement system has also been used for several non-SAFIR projects, and a lot of know-how on e.g. gas-phase and combustion PIV measurements has been acquired. During the last part of 2016, the PIV system will be applied for the 2D/3D velocity field measurements in the PPOOLEX condensation experiments with a sparger in the blowdown pipe mouth. These experiments will continue in 2017, and also high-speed cameras for pattern recognition will be used in these experiments. The PIV system is also planned to be applied to the spray tests in PPOOLEX during 2017.

2.1.2 Evolutionary WMS applications (T1.2)

The advanced applications for the WMS technique have been actively studied in LUT. In 2015, the first results from the axial sensor measurements of two-phase flows were presented in the NURETH-16 conference [Ylönen and Hyvärinen, 2015] and various experiments were conducted to study sensors intrusiveness and performance under different flow conditions. The results were also presented in the SWINTH-2016 workshop in June 2016 [Ylönen et al., 2016]. Example of the results is presented in Figure 2. The participation to the workshop also supported the use of advanced measurement techniques as the workshop is intended for the researchers who are developing and applying these techniques in practice.

Figure 2 Time-averaged void fraction distributions (J_L=1.2 m/s, J_G=0.6 m/s), horizontal to vertical. [Ylönen et al., 2016]

The axial sensor technique and its benefits and possible drawbacks will be further examined in 2017. The applicability of the sensor for measuring phase separation in swirling two-phase flows will be studied further. This will be performed in the HIPE two-phase flow loop test facility by adding a swirling device in the channel. For this purpose, two separate swirling devices with different blade angles were designed and manufactured in 2016. The devices are presented in Figure 3. The first WMS measurements under swirling flow conditions will be conducted by the end of 2016. The data analysis of those measurements will be done in 2017, and more measurements will be conducted if needed.

A new high temperature and pressure WMS has been recently developed at ETH, Switzerland. The applicability of such a sensor for flow measurements in LUT will be studied and a sensor will be acquired to LUT for testing if
found feasible. One possible target of application in LUT could be for example the forthcoming modular test facility (discussed more in the WP3). The new design of the sensor [Kickhofel, 2015] was developed to simplify the manufacturing process of the sensor compared to the previously developed one [Pietruske and Prasser, 2007]. The high temperature and pressure WMS technique was presented also in the SWINTH-2016 workshop, where good knowledge of it was acquired [Kickhofel et al., 2016]. A research visit to ETH Zurich/Helmholtz-Zentrum Dresden-Rossendorf (HZDR) regarding high temperature and pressure WMSs is planned to be made during the first half of 2017.

Figure 3 The swirling devices to be installed in the HIPE test facility.

2.1.3 Improvement of 3D High-Speed Camera data analysis (T1.3)

The improvement of pattern recognition algorithms and related data extraction features have continued in 2016. The CFD modellers are interested in various physical parameters that may be extracted from the high-speed camera footage. The pattern recognition analysis of high-speed camera data is used in development of CFD models for direct contact condensation and interfacial area density calculation. For that purpose, the most important physical parameters obtainable by pattern recognition are interfacial location, velocity, acceleration, bubble volume, surface area and occurrence frequency.

In 2016, the pattern recognition algorithm has been developed further. From high-speed camera measurements in the PPOOLEX test facility it is now possible to calculate volume, surface area, diameter, condensation velocity and acceleration, and (chugging) frequency of condensing bubbles. In 2016, data from the PPOOLEX condensation experiments has been analysed with the algorithm. Figure shows an example of volume, velocity and acceleration of condensing steam bubble calculated from a high-speed camera footage of the PPOOLEX experiment. In 2017, a journal article concerning frequency analysis of chugging condensation with pattern recognition will be written.

By the end of 2016, the analysis of the data from the PPOOLEX condensation experiments with a transparent blowdown pipe by the pattern recognition algorithm will be started. The analysis of the transparent pipe case will continue in 2017. Also data analysis from the condensation experiments with a collar in the blowdown pipe mouth will be started in 2017.
2.1.4 New applications of advanced measuring techniques (T1.4)

The goal of this task is to follow the state-of-the-art of advanced measurement techniques and their applications in thermal hydraulic research. One example of recently developed measurement techniques is the distributed temperature sensor based on Rayleigh-backscatter phenomenon. The sensor utilizing optical fibre enables the measurement of temperature distribution in high detail in different geometries (such as a slab or a rod) [Gerardi et al., 2015]. The distributed temperature sensor technique was presented also in the SWINTH-2016 workshop, where more information concerning it was acquired [Lomperski et al., 2016]. The applicability of the technique to the forthcoming modular test facility will be studied. The task supports the use of advanced techniques in LUT.

2.2 Maintenance and equipment (WP2)

The periodical inspections of the pressure vessels are performed in accordance to the legislation regarding pressure vessels and equipment. Due to the changes in the regulations and the ageing of the equipment, this workload has increased over the years, and cannot be completed with the resources left over from the other experimental work but has to be allocated its own funding.

Aged components will be replaced and the operability of the systems will be ensured by purchasing spare parts to replace broken ones. Typically, these parts include moving parts such as valve and automation components, but also parts for control and data acquisition systems. Since the lifecycle of such products seems to grow ever shorter, a sizeable cache of spare parts is needed in order to minimize the facility downtime.

With the bias of the experimental work moving towards the use of cameras filming underwater occurrences the need for constructing suitable submerged housings for the cameras has risen. The traditional method has been to fit the test vessels with windows for visual observation, but this has led to compromised geometries and problems with the focal lengths. Use of suitable camera housings would rectify these problems, but since the value of a single
camera is in tens of thousands of euros, the construction of the housings is not as simple as it might seem. The margin of error is zero and therefore the investment in design, construction and testing required is quite substantial.

<table>
<thead>
<tr>
<th>Partners in WP2</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>2</td>
</tr>
</tbody>
</table>

2.2.1 PWR PACTEL / PACTEL maintenance (T2.1)

Due to the upgrades in the power grid automation operated by the local power utility, the use of the PACTEL test facility has raised concerns as the unbalanced phase control system used in the core simulator trips the utility’s phase trip alarms. In 2015, the replacement of the core simulator power control equipment was evaluated. The hardware was purchased in 2016.

In 2016, the general maintenance of PACTEL has continued as planned. During the summer hiatus, all of the pressure measurement equipment were calibrated.

In 2017, the maintenance actions of (PWR) PACTEL will be carried out to ensure its availability for the thermal-hydraulic experiments that are planned to be conducted in INTEGRA and INSTAB projects.

2.2.2 Other equipment (T2.2)

In 2015, the procurement process of a computer for the post-processing of computational and experimental data was started, and the computer was ordered in December 2015. The computer was taken into operation in 2016.

In 2016, the upgrade of power transformers has been prepared to increase the electrical power available for the thermal-hydraulic experiments (1 MVA \(\rightarrow\) appr. 2.5 MVA). The upgrade will enable higher heating power to be available for new experimental facilities like “MOTEL.” The higher heating power enables new research topics such as more realistic critical heat flux studies with different fuel geometries. The purchase has been postponed due to a lack of investment decision at LUT.

The electronics needed for temperature measurements with Rayleigh-backscatter principle will be purchased after the specifications for LUT applications have been decided. Applications of the technique are presented in Figure 5. This fibre-optic technique can measure the temperature profile along the length of the fibre at an temperature accuracy comparable to thermocouples and sampling rates < 1 Hz, which is sufficient for most experiments hitherto envisioned. Obtaining a full profile, instead of a few point values, would represent a major advance in providing temperature data for understanding of physics and validation of codes.

The measurement computers (five items) in the laboratory are almost ten years old, and hence they need to be replaced by new computers in 2017.

Figure 5 Different applications of Rayleigh-backscattering principle with optical fibre.
Spare parts inventory needs constant replenishing as the bulk of the instrumentation ages further. The high-speed measurement system will require an upgrade in order to fulfill its purpose in the future test programs.

2.3 Modular Integral Test facility (MOTEL) (WP3)

Different possibilities to construct an integral test facility as a successor of (PWR) PACTEL will be surveyed. This forward planning aims to help to construct a facility without any delays in the future. In 2015, a project proposal was submitted to the Academy of Finland on the topic, and the Academy granted funding for three years to design the concept and build the first parts of the new system. The work package 3 started in 2016 with the study of research based requirements for the new test facility.

<table>
<thead>
<tr>
<th>Partners in WP3</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>2</td>
</tr>
</tbody>
</table>

2.3.1 Research based requirements (T3.1)

Research based requirements for the modular integral test facility were surveyed in 2016. This was conducted on the national level to ensure that the needs of Finnish stakeholders will be fulfilled. In addition, international trends and needs for the thermal-hydraulic experimental research were studied to enable participation to various joint international projects in the future.

A report, which introduces the international trends and future needs in the area of thermal hydraulic research, as well as the research based requirements and guidelines for the design of the new test facility, was written.

2.3.2 Modularity based requirements (T3.2)

Modularity based requirements for the new modular integral test facility will be surveyed in 2017. This will be conducted at LUT to ensure that combining the modules flexibly will be possible.

2.3.3 Preliminary design of MOTEL (T3.3)

No activities in SAFIR in 2017. This activity, and associated hardware activities, could be started early now that supporting Academy funding has also been made available.

2.4 Project management, international co-operation and publications (WP4)

The work package includes the tasks related to the project management and participation to the reference group meetings and the SAFIR2018 midterm seminar. Also international co-operation actions are part of the work package.

<table>
<thead>
<tr>
<th>Partners in WP4</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>LUT</td>
<td>2</td>
</tr>
</tbody>
</table>

2.4.1 Project management and publications (T4.1)

The actions related to the project management, as well as participation and preparation of the SAFIR2018 reference group meetings belong to this task. Preparation of publications (seminar and journal) and dissemination activities are also part of these actions.

2.4.2 SAFIR2018 midterm seminar (T4.2)

Participation and preparation of the SAFIR2018 midterm seminar belong to this task.
2.4.3 International co-operation (T4.3)

The building of expertise on the advanced measurement techniques requires collaboration with other research institutes who are using the same techniques for thermal hydraulic studies. The exchange of experiences on advanced measurements will be continued with institutes such as Paul Scherrer Institut (PSI, Switzerland), ETH (Switzerland) and University of Michigan (U-M, U.S.).

A research visit (2-3 weeks) to U-M (NERS) was made in September/October 2016 regarding WMS measurements and related data analysis methods, PIV and LDV (Laser Doppler Velocimetry) measurements, as well as tomography measurement systems. LUT and U-M have common interests in using custom designed WMSs (as well as PIV measurement systems) for the thermal-hydraulic safety studies [Petrov et al., 2015].

The development of the high temperature/pressure WMS technique at ETH Zurich/HZDR will be followed during 2017. A research visit to ETH/HZDR concerning this issue is planned to be made during the first half of 2017.
## 3. Deliverables and milestones 2017

List of the planned deliverables and milestones for 2017.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1.1</td>
<td>Status report on the advances in thermal-hydraulic measurements (WP1 report)</td>
<td>6</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the data analysis of the WMS measurements of swirling two-phase flow in the HIPE test facility is ready.</td>
<td>N/A</td>
<td>31.5.2017</td>
</tr>
<tr>
<td></td>
<td>- Criterion for approval: the data analysis is done and the results are in a presentable form.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the status report on the advanced measurements is ready.</td>
<td>N/A</td>
<td>31.12.2017</td>
</tr>
<tr>
<td></td>
<td>- Criterion for approval: the status report is written.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>D1.3.1</td>
<td>Journal article: “Frequency analysis of chugging condensation mode in pressure suppression pool systems”</td>
<td>3</td>
<td>30.5.2017</td>
</tr>
<tr>
<td></td>
<td>- The article is planned to be published in International Journal of Heat and Mass Transfer.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the analysis of the data from the PPOOLEX condensation experiments with a transparent blowdown pipe by the pattern recognition algorithm is ready.</td>
<td>N/A</td>
<td>30.4.2017</td>
</tr>
<tr>
<td></td>
<td>- Criterion for approval: the data analysis is done and the results are in a presentable form.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the journal article concerning frequency analysis of chugging condensation is ready for the review process.</td>
<td>N/A</td>
<td>30.5.2017</td>
</tr>
<tr>
<td></td>
<td>- Criterion for approval: the article is written and ready for the review process.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Maintenance of PWR PACTEL / PACTEL</td>
<td>2</td>
<td>31.12.2017</td>
</tr>
<tr>
<td>Milestone</td>
<td>Description</td>
<td>Status</td>
<td>Approval Criteria</td>
</tr>
<tr>
<td>-----------</td>
<td>-------------</td>
<td>--------</td>
<td>------------------</td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the annual maintenance of PACTEL is done.</td>
<td>N/A</td>
<td>maintenance is successfully done.</td>
</tr>
<tr>
<td>D3.1.1</td>
<td>Modularity based requirements of MOTEL report</td>
<td>2</td>
<td>The report describes the main modularity based requirements for the new modular integral test facility.</td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the MOTEL report is written and ready for the internal review process.</td>
<td>N/A</td>
<td>the report is sent to the reviewers.</td>
</tr>
<tr>
<td>D4.2.1</td>
<td>Participation in the SAFIR2018 midterm seminar</td>
<td>0.5</td>
<td></td>
</tr>
<tr>
<td>D4.3.1</td>
<td>Research visit to ETH Zurich in Switzerland/Helmholtz-Zentrum Dresden-Rossendorf (HZDR) in Germany</td>
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<td>A travel report will be written and distributed to the members of the reference group.</td>
</tr>
<tr>
<td>N/A</td>
<td>Milestone: the research visit is done and the travel report is written.</td>
<td>N/A</td>
<td>the research visit is successfully done and the travel report is sent to the RG members.</td>
</tr>
<tr>
<td>N/A</td>
<td>Project management and publications</td>
<td>1</td>
<td></td>
</tr>
</tbody>
</table>

**Total pm** 15
4. Project organisation

Nuclear engineering laboratory and nuclear safety research unit form the project organisation at LUT. LUT is responsible for the whole project. Joonas Telkkä will act as the project manager. This project is planned to be carried out by LUT.

Since this work at LUT is mostly dealing with experiments, it is impossible beforehand to decide the exact working hours of a single person. Thus, only an estimated person months is presented with a full list of the persons who can be involved.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Joonas Telkkä</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP2 WP3, WP4</td>
<td>4.5</td>
</tr>
<tr>
<td>Vesa Riikonen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP2, WP3</td>
<td></td>
</tr>
<tr>
<td>Markku Puustinen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Antti Räsänen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP1, WP2, WP3</td>
<td>1</td>
</tr>
<tr>
<td>Heikki Purhonen</td>
<td>Research director</td>
<td>LUT</td>
<td>WP1, WP3, WP4</td>
<td>0.5</td>
</tr>
<tr>
<td>Virpi Kouhia</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Jani Laine</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Lauri Pyy</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP3</td>
<td>3</td>
</tr>
<tr>
<td>Harri Partanen</td>
<td>Design engineer</td>
<td>LUT</td>
<td>WP2, WP3</td>
<td>1</td>
</tr>
<tr>
<td>Eetu Kotro</td>
<td>Project researcher</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td>1</td>
</tr>
<tr>
<td>Ilkka Saure</td>
<td>Technician</td>
<td>LUT</td>
<td>WP1, WP2</td>
<td></td>
</tr>
<tr>
<td>Elina Hujala</td>
<td>Doctoral student</td>
<td>LUT</td>
<td>WP1</td>
<td>3.5</td>
</tr>
<tr>
<td>Vesa Tanskanen</td>
<td>Post-doctoral researcher</td>
<td>LUT</td>
<td>WP1, WP3</td>
<td></td>
</tr>
<tr>
<td>Heikki Suikkanen</td>
<td>Post-doctoral researcher</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Juhani Vihavainen</td>
<td>Research scientist</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Otso-Pekka Kauppinen</td>
<td>Doctoral student</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Ville Rintala</td>
<td>Doctoral student</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td>Juhani Hyvärinen</td>
<td>Professor</td>
<td>LUT</td>
<td>WP3, WP4</td>
<td>0.5</td>
</tr>
<tr>
<td>Kimmo Tiilinen</td>
<td>Research trainee</td>
<td>LUT</td>
<td>WP2</td>
<td></td>
</tr>
<tr>
<td>N.N.</td>
<td>Research trainee</td>
<td>LUT</td>
<td>WP1</td>
<td></td>
</tr>
<tr>
<td>M.M.</td>
<td>Research trainee</td>
<td>LUT</td>
<td>WP3</td>
<td></td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>15</strong></td>
</tr>
</tbody>
</table>
5. Risk management

Project risks arise from potentially limited availability of human resources, materials, and laboratory work. The primary risks in the project are associated with the unexpected malfunctioning of the research equipment. The repairing of the hardware can take a long time that may lead in significant delays for the planned activities. These risks can be minimized by careful operation of the measurement devices and ensuring that the personnel operating the devices have sufficient knowledge how to operate them.

The foregoing assumes that catastrophic events such as massive fire in the laboratory or LUT campus can be excluded from risk assessment.

There are new users of electric power in LUT laboratories, and it means that the scheduling of the experiments is now more challenging.
References


Smith, B. L., 2015. CFD validation experiments: What’s the difference?. Proceedings of NURETH-16, Chicago, U.S.


### Resource Plan for 2017

#### Annex 2-1

**Liite 2-1**

Development of thermal-hydraulic infrastructure at LUT

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Volume</td>
<td>Personnel</td>
</tr>
<tr>
<td></td>
<td>person</td>
<td>keuro</td>
</tr>
<tr>
<td>WP1 - Advanced measurement techniques</td>
<td>9.0</td>
<td>102</td>
</tr>
<tr>
<td>T1.1 Advanced and combined use of PIV/WMS/3D Cam systems</td>
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<td>28</td>
</tr>
<tr>
<td>T1.2 Evolutionary WMS applications</td>
<td>2.5</td>
<td>28</td>
</tr>
<tr>
<td>T1.3 Improvement of 3D High-Speed Camera data analysis</td>
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<td>34</td>
</tr>
<tr>
<td>T1.4 New applications of advanced measuring techniques</td>
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<tr>
<td>WP2 - Maintenance and equipment</td>
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</tr>
<tr>
<td>T2.1 PWR PACTEL/PACTEL maintenance</td>
<td>1.0</td>
<td>10</td>
</tr>
<tr>
<td>T2.2 Other equipment*</td>
<td>1.0</td>
<td>10</td>
</tr>
<tr>
<td>WP3 - Modular Integral Test facility (MOTEL)</td>
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<td>22</td>
</tr>
<tr>
<td>T3.1 Research based requirements</td>
<td></td>
<td></td>
</tr>
<tr>
<td>T3.2 Modularity based requirements</td>
<td></td>
<td></td>
</tr>
<tr>
<td>T3.3 Preliminary design of MOTEL</td>
<td></td>
<td></td>
</tr>
<tr>
<td>WP4 - 2.4.1 Project management, international co-operation and publications</td>
<td>2.0</td>
<td>23</td>
</tr>
<tr>
<td>T4.1 Project management and publications</td>
<td>1.0</td>
<td>11</td>
</tr>
<tr>
<td>T4.2 SAFIR2018 midterm seminar</td>
<td>0.5</td>
<td>6</td>
</tr>
<tr>
<td>T4.3 International co-operation**</td>
<td>0.5</td>
<td>6</td>
</tr>
<tr>
<td>TOTAL</td>
<td>15.0</td>
<td>167</td>
</tr>
</tbody>
</table>

**Comments:**

Travel costs in WP1 and WP4 include international activities such as research visit to ETH Zurich (Switzerland)/Helmholtz-Zentrum Dresden-Rossendorf (Germany)

* Purchase of the electronics of optical fibre temperature measurement application, replacement of data acquisition and control system computers

** Visits to HZDR in Rossendorf and ETH.
SAFIR2018 Project plan

JHR

JHR collaboration & Melodie follow-up

Santtu Huotilainen, Ville Tulkki, Petri Kinnunen
VTT Technical Research Centre of Finland Ltd
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   1.2 Objectives and expected results .........................................................4
   1.3 Exploitation of the results ...............................................................4
   1.4 Appropriateness of the project to SAFIR2018 programme ............... 5
   1.5 Education of experts ................................................................. 5

2. Work plan .............................................................................................6
   2.1 Work package 1 (WP1) – JHR collaboration ..................................... 6
       2.1.1 Task 1 (T1.1) – WG participation ............................................ 6

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4. Project organisation .............................................................................8

5. Risk management .............................................................................9

References ............................................................................................10
1. Research theme and motivation

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

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

The delivery of the Melodie device was part of the Finnish in-kind contribution, and in this part the work was successfully completed in 2012. The Melodie in-core experiment, carried out in 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. During the experiment, valuable irradiation creep data was also produced. The second WP of this project focused on following the Melodie experiment and its possible continuation project, and bringing the knowledge on the feasibility of the technology as well as the data back to VTT and the SAFIR2018 community. This WP ended in 2016.

1.1 Background and state-of-the-art

European MTRs have provided an essential support for nuclear power programs over the last 40 years within Europe. However, these MTRs will be more than 50 years old this decade, and they will face an increasing probability of shutdown due to the obsolescence of their safety standards and of their experimental capability. It is acknowledged that in the coming decades the JHR, a new MTR currently under construction at CEA Cadarache research centre, will function as a major research reactor within Europe for nuclear energy. JHR will represent an essential research infrastructure for scientific studies dealing with material and fuel behaviour under irradiation. The reactor will perform R&D programs for the optimisation of the present generation of nuclear power plants (NPP), support the development of the next generation of NPPs (mainly light water reactors, LWR), and also offer irradiation capabilities for material testing for future reactors.

JHR will offer modern irradiation experimental capabilities to study material & fuel behaviour under irradiation. JHR will be a flexible experimental infrastructure to meet industrial and public needs within the European Union related to present and future Nuclear Power Reactors. JHR is designed to provide a high neutron flux (twice as large as the maximum available today in MTRs), to run highly instrumented experiments to support advanced modelling giving prediction beyond experimental points, and to operate experimental devices giving environmental conditions such as pressure, temperature, flux, and coolant chemistry relevant for example for water reactors, for gas cooled thermal or fast reactors, and for sodium fast reactors. These objectives require representative tests of structural materials and fuel components, as well as in-depth investigations with separate-effect experiments, coupled with advanced modelling.

According to the consortium agreement, JHR is aimed to become an international user facility with the model of the Halden Reactor Project with multinational projects and proprietary experiments. Consequently, CEA is preparing, with the support of the OECD/NEA, a joint programme, which has the strategic scope to address fuel and material issues of common interest that are key for operating and future NPPs (mainly focused on LWR).
Even though the construction of the reactor is still in progress, the planning of JHR experimental devices and the determination of experimental needs has been already started. The planning is done mainly by three working groups with representatives from all members of the international consortium.

The first experimental programmes will be planned and carried out within the H2020 project FIJHOP (Foundation for future International Jules HOrowitz experimental Programs), applied for in the fall 2016 Euratom call. The project includes fuel and material irradiation experiments, which will be performed in existing European MTRs. Furthermore, a post-irradiation examination (PIE) program as well as modelling for both fuel and materials will be included. The total number of 20 participants in the FIJHOP project consists of JHR consortium members and other European industrial partners, research institutes and universities.

### 1.2 Objectives and expected results

According to the current schedule the construction of the JHR will be ready in 2021 and the first experiments will start in 2022. The planning of these experiments has already been launched within three WGs, namely Fuel WG, Materials WG, and Technology WG, aiming first to a pre-JHR programme carried out in the existing European nuclear facilities, and through this work aiming at clarifying the experimental parameters and conditions needed for the JHR. The objectives of these working groups 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 European research project FIJHOP, gathering a total of 20 partners including the 12 from the JHR consortium, is structured on the two following scientific challenges identified by the JHR Working Groups on Fuel and Material R&D topics:

- FIJHOP-F studying the quantification of phenomena activated in an LWR fuel rod during power transients with focus on those impacting cladding loading and limiting core management in a power reactor;
- FIJHOP-M investigating the neutron spectrum effects on the degradation of low alloy steels (RPV) and stainless steels (reactor internals), and the dose-damage relationship quantified by microstructural characterization and mechanical testing.

Considering the well-integrated MTRs network within Europe, it has been acknowledged by the JHR International Consortium as well as by the NUGENIA community that there is particular importance to prepare, by using today’s operating MTRs, future joint international programs that will be performed in JHR. This is the main general objective of the FIJHOP proposal.

The participation in the three working groups 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.

### 1.3 Exploitation of the results

The information on the experimental capacity of JHR acquired in the WGs can be used as a guideline in the planning of new experiments. The WGs will assist the JHR research programme management in planning and realisation of the experimental campaigns. In addition to bringing out our own interests and needs, the WG discussion about general experimental needs and possibilities is useful when making decisions about the participation and collaboration in the future programmes. Furthermore, these needs can be used in the development of new experimental devices in the on-going design phase, which further helps to plan appropriate experiments. The findings and results of WG work will be available immediately, and the information will be specified and expanded as the work progresses.
1.4 Appropriateness of the project to SAFIR2018 programme

The planning and designing of the future experiments in JHR is a topical issue, and Finland has committed to participate in this international collaboration as part of the nuclear safety related research activities. JHR will expand the possibilities in this field in a way that would not be possible nationally. In Finland the SAFIR programme is the most suitable community to participate and disseminate the JHR experimental information. In the future, the research activities in the JHR will be closely linked to the VTT Centre for Nuclear Safety, where post-irradiation examination of irradiated materials can be carried out. Therefore, active participation in the preparation phase of the JHR project is important in helping create links between the international research activities and our domestic efforts, establish partnerships for future collaboration, and develop our competences and infrastructure to be able to meet the national requirements for adequate knowledge and training in nuclear safety.

1.5 Education of experts

This project aims at active participation in the three JHR WGs. Each WG has planned to meet twice a year. Educational point comes from the participation in planning the experimental campaigns for JHR. The planning work will be done in close cooperation with the other JHR consortium member organisations. For example, the role of the Technology WG is not only to plan the technological choices for the future tests but also make cost-benefit estimates for the tests. This kind of work has not been done before at VTT.

This project will not aim at producing theses or dissertations.
2. Work plan

2.1 Work package 1 (WP1) – JHR collaboration

The work plan includes participation in the international JHR WG meetings in all three WGs as well as in the annual JHR Technical Seminar. The working groups will meet at least twice a year in 2015-2018. The annual JHR Technical Seminar will be arranged in Cadarache every spring. In the previous seminars the progress of the JHR construction project has been presented, and each member organisation have also presented their in-kind status and each WG their meeting outcomes. All the future JHR owners are represented in the seminar. In 2016, the seminar was combined with the Nugenia Forum and held in Marseille, in an effort to attract interest towards the planned pre-JHR experiments also among non-consortium members. These pre-JHR experiments for LWR fuel and materials will be carried out in the FIJHOP project within the H2020 programme during 2017-2022. The next JHR technical seminar will be combined with the FIJHOP kick-off meeting in the case of a successful project application.

In 2017, an important part of the WG work will be the further elaboration and planning of the objectives of the FIJHOP project, in terms of preparation for and comparability with the future JHR irradiation experiments. Through the FIJHOP project, the consortium members also have the opportunity to practice and improve the international co-operation considering active material transport and PIE in different countries. For VTT and the Finnish nuclear community this is an important step towards starting the active materials research activities in the Centre for Nuclear Safety by getting an opportunity to train our staff in the active materials handling and the use of the new hot cells through international project work, and by developing the work practices and processes for more efficient use of the facilities.

In 2017, one WG meeting will be organised in Finland by VTT. In this meeting the Finnish utilities have a great opportunity to get updated information on the JHR and the future experimental programmes. Furthermore, it is a good opportunity for us to present and commercialise our new Centre for Nuclear Safety to other JHR consortium members, regarding our participation in the materials PIE in the future experimental programmes.

<table>
<thead>
<tr>
<th>Partners in WP1</th>
<th>Person months</th>
</tr>
</thead>
<tbody>
<tr>
<td>VTT</td>
<td>1.7</td>
</tr>
</tbody>
</table>

2.1.1 Task 1 (T1.1) – WG participation

In 2017, the members of the project team will participate in WG meetings. The goal is to participate in the planning work of the WGs and enable a bidirectional exchange of information. The meetings are held in spring and usually also in fall. Furthermore, a JHR Technical Seminar is envisioned in June 2017. In the case of a successful FIJHOP project application, the members of the project consortium will be invited to the JHR seminar and the FIJHOP kick-off meeting will be organised in conjunction with the seminar.

The second WG meeting of 2017 will be held in Finland in the fall. VTT will make the arrangements for the two-day meeting, held in the Centre for Nuclear Safety. The meeting will include a tour of the active materials research laboratory, where hot cells will have been installed.

The main objectives of the meetings are to find a consensus among the consortium about the detailed specifications of the FIJHOP irradiations and post-irradiation examinations, and to establish what are the next steps towards the first experimental programmes in JHR. The information gathered from these meetings will be disseminated to the SAFIR2018 community and the continuous feedback from Finnish stakeholders is in turn communicated to the WGs.
3. **Deliverables and milestones 2017**

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable or milestone name</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
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<tr>
<td>D1.1.1</td>
<td>Travel report from the WG meetings and JHR technical seminar</td>
<td>1.0</td>
<td>31.5.2017</td>
</tr>
<tr>
<td>D1.1.2</td>
<td><strong>WG meeting in Finland</strong>&lt;br&gt;Criterion: meeting held successfully during 2017</td>
<td>0.7</td>
<td>15.12.2017</td>
</tr>
</tbody>
</table>

| Total pm           | 1.7                                                                                         |

*Short description of the deliverable or milestone. Criterion for the approval of the milestone.*
4. Project organisation

The project organisation is listed in the table below. Santtu Huotilainen is the project manager.

<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2017)</th>
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</thead>
<tbody>
<tr>
<td>Santtu Huotilainen</td>
<td>Research scientist</td>
<td>VTT</td>
<td>1.1</td>
<td>1.1</td>
</tr>
<tr>
<td>Petri Kinnunen</td>
<td>Research manager</td>
<td>VTT</td>
<td>1.1</td>
<td>0.3</td>
</tr>
<tr>
<td>Ville Tulkki</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>1.1</td>
<td>0.3</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td></td>
<td><strong>1.7</strong></td>
</tr>
</tbody>
</table>
5. Risk management

There are no risks foreseen in the project at the moment. In the case of possible changes in the WG representation, the continuation of knowledge transfer would have to be ensured.
References


JHR collaboration & Melodie follow-up

<table>
<thead>
<tr>
<th>Work packages and Tasks</th>
<th>Volume</th>
<th>Personnel</th>
<th>Mat &amp; supp</th>
<th>Travel</th>
<th>Ext serv</th>
<th>Memb fee</th>
<th>Other</th>
<th>TOTAL</th>
<th>VYR</th>
<th>VTT</th>
<th>Other</th>
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</thead>
<tbody>
<tr>
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<td>0.0</td>
<td>7.8</td>
<td>0.0</td>
<td>0.0</td>
<td>29.0</td>
<td>20.0</td>
<td>9.0</td>
<td>0.0</td>
<td></td>
</tr>
<tr>
<td>T1.1 WG participation</td>
<td>1.7</td>
<td>21.2</td>
<td>0.0</td>
<td>7.8</td>
<td>0.0</td>
<td>0.0</td>
<td>29.0</td>
<td>20.0</td>
<td>9.0</td>
<td>0.0</td>
<td></td>
</tr>
<tr>
<td>TOTAL</td>
<td>1.7</td>
<td>21.2</td>
<td>0.0</td>
<td>7.8</td>
<td>0.0</td>
<td>0.0</td>
<td>29.0</td>
<td>20.0</td>
<td>9.0</td>
<td>0.0</td>
<td></td>
</tr>
</tbody>
</table>

Comments:
SAFIR2018 Project plan

RADLAB
Radiological laboratory commissioning, 2017

Wade Karlsen
VTT Technical Research Centre of Finland Ltd

Seppo Tähtinen
VTT Technical Research Centre of Finland Ltd
PREFACE

Pursuant to the letters of invitation by the Ministry of Economic Affairs and Employment for SAFIR2018 (TEM/1236/08.09.01/2016) and KYT2018 (TEM/1403/08.09.02/2016), VYR is prepared to support the renewal of the radiological laboratory research infrastructure via three instruments: 1) the research and infrastructure instrument, generally comprised mainly of personnel, travel and associated research execution expenses; 2) a special allocation for supporting the VTT Ltd Centre for Nuclear Safety radiological laboratory facility expense, and 3) a special allocation for supporting the VTT Ltd Centre for Nuclear Safety radiological laboratory equipment investment expenses. All three instruments are jointly supported by SAFIR2018 and KYT2018, and therefore this is joint proposal for funding from both programmes.

The utilization of all three instruments are foreseen to be coordinated by the RADLAB project, as that is the only project with allocations primarily for personnel salaries, while the latter two funding instruments are aimed at facility costs and direct equipment investment costs, respectively. Thus, the execution of the project portfolio proceeds such that RADLAB carries out the work associated with the tasks, but the facility costs of the radiological laboratory are supported directly by RADCNS, while particular equipment investment costs are actually paid from the RADINFRA project. The applied funding includes appropriate VAT based on the law and reasoning of “Laki ydinenergialain muuttamisesta (HE320/2014).” The investment-related deliverables are shared deliverables between RADLAB and RADINFRA. Because the equipment investments will exceed the maximum amount possible to apply for in a given year, subsequent years will claim the costs retroactively.

The revised breakdown of funding utilization for 2017 is as follows:

<table>
<thead>
<tr>
<th></th>
<th>SAFIR2018, euros</th>
<th>KYT2018, euros</th>
<th>TOTAL, euros</th>
</tr>
</thead>
<tbody>
<tr>
<td>Implementation project (RADLAB)</td>
<td>350,000</td>
<td>143,000</td>
<td>493,000</td>
</tr>
<tr>
<td>VTT CNS laboratory facility (RADCNS)</td>
<td>2,050,000</td>
<td>650,000</td>
<td>2,700,000</td>
</tr>
<tr>
<td>VTT CNS equipment investments (RADINFRA)</td>
<td>2,740,000</td>
<td>860,000</td>
<td>3,600,000</td>
</tr>
</tbody>
</table>

This project proposal is focused on the RADLAB project.
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1. Research theme and motivation

In this third year of the SAFIR2018 research programme, the technical commissioning of the VTT Centre for Nuclear Safety (CNS) will be completed as the hot cells are installed, and the final radiological commissioning will be well underway. The VTT CNS and its hot cell facility is a national infrastructure hosted by VTT, and is considered an important element in fulfilling the national requirements for independent competencies for domestic nuclear power generation.

This RADLAB project was preceded by the REHOT project in the SAFIR 2014 program and in the first year of the SAFIR 2018 program. The RADLAB project is an integral part of the overall infrastructure renewal process surrounding the VTT CNS, in support of both reactor safety and nuclear waste management (NWM) research. While the former REHOT project focused mainly on the design, construction and equipping of the new CNS facilities, as shown in the schematic of Figure 1, this RADLAB project spans the move from the existing facilities at Otakaari 3 (OK3), to the new facilities, and features the commissioning of equipment and ramping up of the infrastructure in the new facilities. In 2017 there is a special emphasis on nuclear waste management research infrastructure commissioning, expanding the high resolution radioanalytical readiness and materials testing capabilities of the CNS, and also renewing and improving the condition monitoring of the experiments in aerobic and anaerobic experiment environments.

The RADLAB project involves efforts in five main areas: 1) hot cell fabrication, installation and commissioning; 2) hot laboratory equipment procurement and nuclearization; 3) design, fabrication and installation of self-built research facilities; 4) design, fabrication and installation of materials handling and storage facilities; and finally, 5) management of the full laboratory infrastructure commissioning and ramp-up of operations for both reactor safety and final repository research.

Figure 1: Radiological laboratory infrastructure renewal process comprised of simultaneous decommissioning of facilities at Otakaari 3 (OK3) and equipping and commissioning of the Centre for Nuclear Safety.
1.1 Background and state-of-the-art

VTT has been hosting the Finnish national radiological laboratory infrastructure since the first nuclear power plants were constructed in Finland in the 1970’s. Historically the principle radioactive materials handling has been for the testing of reactor pressure vessel steels, but over time the activities have broadened to outgrow both the capacity and capabilities of the existing facilities. As such, a decision was made in 2011 to build a whole new facility, with the additional goal of gathering most of the VTT Nuclear Safety research personnel scattered around the Otaniemi campus, into a single, compact facility called the VTT Centre for Nuclear Safety (CNS). The facility would house new radiological laboratories and contemporary hot cells for reactor safety and nuclear waste management research.

The design of the VTT CNS in its current rendition has been on-going since 2012 in the REHOT project. The process was initiated in 2008 with preliminary assessment of the needs and options as a part of the SAFIR2010 AKTUS project. The facility is owned by the Finnish State real-estate company Senaattikiinteistöt, with VTT renting the building from them as the end user. As such, the facility design process was carried out in close cooperation between VTT researchers (the end users) and the design team employed by Senaattikiinteistöt. Regulation and oversight by the authorities has involved the local municipal government, building department and emergency services, as well as Radiation and Nuclear Safety Authority, Finland–STUK.

The location of the new VTT CNS is Kivimiehentie 3 in Otaniemi. The location and position is shown in Figure 2. Since in some cases shipment of radioactive materials involves large trucks, the location enables driving and turning access for large tractor and trailer combinations. As visible in the site layout (Figure 2), the facility includes an office wing and a laboratory wing nestled amongst the trees. The office wing is 3,300 m² and includes a ground-level conference centre, above which are three floors of modern, flexible office space for 150 people. It features an architecturally striking, angular facade on the Kivimiehentie street side, producing the distinct, yet complementary appearance shown in Figure 2, blending in with the natural surroundings. The office wing houses nuclear sector researchers in areas such as computerized fluid dynamics, process modelling (APROS), fusion plasma computations, severe accidents, core-computations, nuclear waste-management and safety assessments, as well as the staff working in the laboratory wing. The laboratory wing is a more conventional, rectangular wing and includes a basement level and two floors of laboratory space. The laboratory activities include research involving radiochemistry, nuclear waste management, dosimetry, failure analysis as well as mechanical and microstructural characterisation of structural materials. Shipping radioactive materials into and out of the facilities occurs through a gated courtyard and covered loading dock at the basement level, at the rear of the building.

![Figure 2: VTT Centre for Nuclear Safety in Otaniemi.](image)

Over the course of the SAFIR2014 program, the renewal of the hot cell infrastructure made great strides forward in the REHOT project. Based on the assessment of needs conducted in the AKTUS project, a draft facility design was made in collaboration between VTT and A-Insinöörit, the engineering company employed by Senaattikiinteistöt. This was subsequently turned into an engineering design during 2012, and over the 2013 period the floor plans of
the facility were subjected to some further modifications, and then the detailed design process was carried out to generate the detailed layouts of furnishings and building systems on a room-by-room basis. In 2013 the draft plans of the laboratory facility and the activities it is foreseen to contain, as well as the practices that will be employed, were presented to STUK’s radioactive facilities evaluators for review. A preliminary, non-binding assessment was received with some specific recommendations for improvement, which were subsequently implemented into the design.

With regards to the hot cells themselves, a significant milestone in 2013 was the execution of a hot-cell conceptual design and preliminary cost estimate, which was made on a contract with Merrick & Company. A reassessment of the design was then carried out by VTT and utilized for the tendering process for the engineering design and fabrication of the hot cell facilities, carried out in 2014. The contact was awarded to Isotope Technologies Dresden GmbH (ITD).

The ground breaking for the CNS began immediately in 2014, so during the SAFIR2018 programme the CNS progresses through the construction and commissioning stage of the radiological laboratories, including the hot cell facilities. Although the renewal of the research environment does not provide time for significantly modifying the goals set for the programme, it creates new possibilities for research with its new hot cell and laboratory facilities and equipment needed for research on reactor structural materials and nuclear waste management. VTT took delivery of the laboratory in May 2016, and installation of the hot cells will take place in the first half of 2017.

The CNS and its hot cell facilities enable progress beyond the state-of the art on many fronts. The modern laboratory facilities provided by the CNS raises the level of technological prowess, in tandem with enhancing radiation safety. By bringing most of the VTT nuclear sector researchers together under the same roof, the capacity to act together in wide-ranging national programs is enhanced. Likewise, the CNS provides a common radiological facility housing radiochemistry laboratories, modern microscopy and analytical capabilities, and mechanical testing of radioactive materials. This offers a better platform for national multidisciplinary collaboration.

The technological enhancement enables a higher scientific level in research results. This in turn is essential for participation in international research projects and programmes, as well as for enabling high-level research in support of doctoral programs. The enhanced radiation safety offered is also beneficial for providing an environment amenable to hosting visiting researchers, whether as part of domestic degree programs, or of international consortium research projects.

With respect to the hot cells themselves, the greatest change is the essential improvement of research capacity for highly-active, neutron-irradiated, austenitic stainless steels and freshly irradiated pressure vessels steels. And as a national research infrastructure coupled with Finland’s share of the Jules Horowitz Material Test Reactor (JHR), for the first time in the history of Finnish nuclear R&D, the country has the possibility to carry out the whole material testing chain, starting from neutron irradiation in the JHR core, through to comprehensive mechanical, microstructural and analytical post-irradiation examinations (PIE) in the CNS radiological facilities.

1.2 Objectives and expected results

The infrastructure for radioactive materials research and testing involves facilities, equipment and competent users. The RADLAB project is the means by which the infrastructure investments are executed, supporting the personnel involved in carrying out the work. This includes design input and oversight of ITD in designing and manufacturing the hot cells, but also in training of personnel, adopting a new safety culture, executing the key equipment procurement processes, nuclearization of equipment going into the cells, and the design, procurement and installation of the other research devices and process equipment supporting the radioactive materials handling and storage.

In 2017 the researchers from the various disciplines are sharing the same office building and the new nuclear waste management and radiochemistry laboratory facilities are already functional. Thus, the focus can turn to exploiting the modern equipment and tools to ensure continued high-quality nuclear waste research, while also diversifying the methods for a broader and fuller chain of analytical services.

The ultimate objective of the hot cell contract with ITD is to achieve safe, functional hot cells in a cost effective manner, which are appropriate to the specified research and testing needs. The detailed design was completed on schedule at the end of 2015, and manufacturing of the units has taken place in 2016. In accordance with the current work plan laid out by ITD (Appendix A of this document), installation and technical commissioning of the hot cells at VTT is on track to occur mainly during the first half of 2017.

The primary objective of the equipment procurement is to acquire the most suitable and cost effective hot laboratory, hot cell and ancillary devices and instruments for the specified needs. The end result is therefore delivery (and payment) of each purchased piece of equipment, and demonstrated functionality of each self-built device. However, the process for each piece of purchased equipment involves carrying out the multistage competitive bid process involving information gathering, technical specification development, supplier identification, tendering, ordering, factory acceptance testing, and then taking delivery and commissioning of the device. Further, for the entire
palette of equipment, the installation and payment schedules must be synchronized with the sources of payment, which come from both VYR and VTT. The procurement schedule is constantly updated as more information is discovered on the needs, suppliers and alternatives. **The current procurement schedule for ALL of the devices and equipment, both purchased and self-built, is shown in Appendix B of this document.**

The overall objective of the equipment nuclearization and installation (whether purchased or self-built), is to achieve safe functionality of the device in its application for radioactive material handling or testing, whether it is a self-built “hot” autoclave system, a stand-alone device like a “hot” SEM, or a device deployed inside one of the hot cell chambers. In the case of the hot cell suite manufactured by ITD, a full Factory Acceptance Test (FAT) and Site Acceptance Test (SAT) procedure is specified in the contract.

In a general sense the commissioning of the new equipment involves the main phases of installation, functionality testing, and commissioning before utilization. In the case of new devices, the commissioning is the final phase of the procurement, but even in the case of existing devices that are moved to the new facilities, the same elements are relevant. As shown in the schematic of Figure 3, each of these main phases can involve several steps.

![Figure 3. Main stages of device commissioning.](image)

**1.3 Exploitation of the results**

While the principle goal of the RADLAB project is to execute the infrastructure renewal, the exploitation of that result is firstly achieved by **demonstration of the functionality** of the facilities for producing mechanical and microstructural data and results of radioactive materials in conditions that are in line with the ALARA principle expected of contemporary radiological facilities. The initial exploitation is by the complementary European commission funded research project SOTERIA, the Academy of Finland funded MENUCHAR project and the BREDA project. The goal of the programme period will be to demonstrate the research capacity of the facility, for use in increasing the overall understanding of the effect of radiation on nuclear power plant structural materials. The project proposals H2020-LOTO (on material harvested from a decommissioned NPP) and SA-ATF (on accident tolerant fuel claddings) will also utilize the new testing capability if funded. The primary areas of research involve life cycle extension, reactor pressure vessel steels, austenitic stainless steels (internal components), but as new topics there is irradiated fuel element structural parts and fuel cladding materials (GEN III–IV), and consideration also of capabilities to research irradiated concrete. Source material for such tests and examinations should ideally be from materials harvested from NPPs, both domestic and international. With particular regards to testing and characterizing materials from in-core irradiations and tests, the Halden Reactor Project provides materials from real plants, and the Jules Horowitz Reactor project is followed in collaboration with a separate project in SAFIR2018. By the end of 2018 the overall goal is to have demonstrated functionality of the new facilities, and in doing so, added to the overall understanding of radiation effects on RPV, internals and fuel cladding materials performance. The relationship between the infrastructure projects RADLAB and JHR and other research projects like MENUCHAR and EU-SOTERIA is illustrated Figure 4.
On another level, the VTT CNS gathers much of the VTT Nuclear Safety research personnel earlier scattered in Otaniemi, into a single, compact location. This will promote synergism between researchers across topics and facilitate closer collaboration between experimental and modelling work. A first tangible step in this direction is collaboration between the JHR and PANCHO projects realized in SAFIR2018, in which biaxial creep experiments on fuel cladding material in JHR are joined with cladding creep modelling work in PANCHO, which involves VTT researchers that are housed in the office wing of the VTT CNS.

The proximity of the VTT CNS to Aalto University will also help to strengthen the symbiotic relationship that exists between the two institutes. The existence of shared facilities and equipment will promote the synergy of Finnish research on existing and future reactor concepts. A concrete first example of this is the MENUCHAR project, in which Aalto University will develop a “hot” positron annihilation spectroscopy device for deployment in the new facilities, initially for use in evaluating point defect populations in irradiated RPV steel.

Finally, modern radiological research facilities strengthen Finland’s capacity and capability to contribute to international projects as an equal partner or even leader.

1.4 Appropriateness to SAFIR2018/KYT2018 programme

The VTT Centre for Nuclear Safety has been identified as one of the key development areas in the SAFIR2018 programme. The research to be carried out in the CNS is on GEN II–IV reactors, fusion energy and nuclear waste management.

The hot cell facilities are a key feature of the CNS for reactor safety research, since the principle use of the hot cells is in the assessment of durability, integrity, failure and damage mechanisms of reactor materials, particularly those contaminated or activated materials of the primary circuit. This helps generate knowledge and data that can help in modelling of aging and failure mechanisms of components in nuclear power plants (both physical and chemical), particularly regarding the assessment methods for the radiation tolerance of materials and the effect of radiation on long-term characteristics. It furthers the infrastructure capability and capacity to a higher level of safety even while allowing materials with a higher level of radioactivity to be researched, like neutron irradiated stainless steels of reactor internals components.

Likewise, about half of the laboratory wing of the CNS is currently dedicated to radiochemistry and bentonite research for nuclear waste management. For example, the clean-room facilities with the high resolution mass spectrometer enables measuring low concentrations of uranium isotopes in solubility studies, boron isotopic ratios in process waters and low levels of Si, Al, Fe and Mg in bentonite waters. Likewise, new liquid scintillation counter equipment in the facility would be used not only for routine dosimetry applications, but also to analyse C-14 in the...
nuclear power plant decommissioning waste dissolution studies, and to analyse radium in nuclear waste management studies.

Valuable input from the power companies in the facility utilization is solicited in an ad hoc forum addressing, among other things, perspective on specific technical solutions, building and strengthening the network with other international experts in irradiated materials research and testing, and ultimately in foreseen relevant research needs coming in the future. For 2016 this took the form of a Centre for Nuclear Safety steering group based on a Memorandum of Understanding signed in 2015 between VTT Oy, the Ministry of Employment and the Economy, and the Finnish nuclear power plant operators. Now that the laboratory and hot cells are being commissioned for research and testing work, the intention for 2017 is to have a User’s Group that will facilitate the utilization of the facilities for the mutual benefit of the Finnish stakeholders.

1.5 Education of experts

A key component of the infrastructure for radioactive materials research and testing is competent users. As such, education of experts in the RADLAB project is mainly comprised of training the users of the equipment procured for the facility, such that they can operate them proficiently and purposefully. This includes not only those devices installed in the laboratory facilities, but the hot cells themselves and their associated devices, in particular as it relates to assuring a healthy safety culture. This mainly involves hands-on training in the remote handling theatre, and as such, no theses or dissertations are expected within this project. Likewise, participation in courses and conferences related to the research topics improves the purposefulness of the equipment specification and utilization. In particular, participation in the HOTLAB conference series continues to be a fruitful way to educate experts on remote handling and other technical solutions concerning working in hot cells.

As the new devices in radiochemistry become useable in their new laboratory environment, there is particular need in 2017 for development and training for use of the HR ICP-MS (High Resolution Inductively Coupled Mass Spectrometer) and radiation spectrometers like the new liquid scintillation counter (LSC).
2. Work plan

The project is executed in 5 main work packages, each including a number of tasks, as shown in Figure 5. The first one focuses on the hot cell fabrication, installation and commissioning process, which involves not only participation in the execution of the hot cell contract with ITD, but also continued development of the hot cell functions and training of operators for the hot cell facilities. In the second work package the hot laboratory equipment is procured, which involves not only executing the procurement process for each device, but also managing the overall investment schedule. The third work package focuses on self-built research facilities, including their design, fabrication and installation, as well as device adaptation for use in the hot cell theatre with appropriate modifications for e.g. remote operation and radiation protections (“nuclearization”). In the fourth work package the facilities for handling and storage of radioactive materials and waste are designed, fabrication and installed. Finally, in light of the integral nature of this work with the realization of the VTT CNS, there is a fifth work package that focuses mainly on the organization of the VTT CNS, commissioning, and ramp-up of the infrastructure utilization.

2.1 Hot cells (WP1)

Over the 2016-2018 period the hot cells work package involves executing the hot cell contract with ITD, as well as developing the hot cell functions and training of personnel at VTT for the new hot cell environment. The hot cells are procured as a sub-contract with ITD according to the documentation laid out in the tender executed in 2014. The conceptual design made with ITD was described in a report prepared during the REHOT 2015 project [1]. The engineering design was mainly executed in 2015, and most of the manufacturing took place in 2016. The installation and commissioning of the hot cells is the primary activity in 2017. The manufacturing, installation and commissioning of the hot cells in the new CNS requires participation by VTT personnel in many aspects. These design and oversight activities are carried out in this work package. As laid out in Appendix A, installation and commissioning of the hot cell units is carried out mainly in the first part of 2017. During this period VTT reviews quality assurance reports, evaluates manufacturing-related documentation, and familiarizes themselves with the construction and operation of the cells while overseeing the installation in the CNS. This is concluded with the Site Assessment Testing, and followed up by integration of the remaining devices in the cells.

Simultaneous to the manufacturing and installation, the personnel utilizing the hot cell facilities for materials testing and research continue to develop their skills in the new ways of working that the new facilities will offer. This entails not only significantly more remote handling skills via manipulation devices and development of semi-automation, but more generally the development of methods that can reduce the amount of radioactive waste produced, reduce the overall volume of radioactive material that needs to be handled at a given time, and overall promotion of the safety culture of the operators and project managers alike. This is accomplished by offering exchange opportunities for operators at other hot cells in Europe, attending the HOTLAB and other user-focused conferences and meetings relevant to the skills development, and also formal training by ITD.

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<tr>
<th>Partners in WP1</th>
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<td>VTT</td>
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2.1.1 Hot cell installation by ITD (T1.1)

The expectation is that most of the manufacturing of the new hot cells will be completed by the end of 2016, and installation will mainly take place in the first part of 2017. They are manufactured in a series of Cell Group units, each having a separate Factory Acceptance Test (FAT). There are a total of 5 Cell Groups: 1A (hot cells 1.3, 1.6 & 1.7), 1B (hot cells 1.1 & 1.2), 1C (hot cells 1.4 & 1.5), 2 (light cell) and 3 (basement cell). Most FATs are expected to be completed by the end of 2016, with 1C in February 2017. Following the FAT of each Cell Group, installation proceeds in the following main steps:

- Disassembly / Packing / Shipment of Cell Group
- Equipment move-in of Cell Group
- Installation & Pre-Commissioning of Cell Group
- Site Acceptance Test
- Training

In 2016 weekly meetings were held between VTT and ITD by teleconference, and there were about 8 face-to-face meetings, most often at the ITD factory. In 2017 the work will mainly take place at VTT, so travelling to ITD’s factory in Germany is expected to be much less than that in 2016.

In 2015 the QA and documentation oversight at VTT was organized by launching a sub-contract with Qualifinn Oy for evaluating the Engineering Design Quality Plan presented by ITD, and this work continued in 2016 for evaluating the Manufacturing Quality Plan of ITD. In 2017 the QA follow-up will focus on the documentation review related to the Site Acceptance Plan, and it is expected that Qualifinn Oy will continue to aid VTT in this work.

2.1.2 Training (T1.2)

To make use of the infrastructure, there is a need to establish a safe, effective and efficient radiological workforce meeting the research and testing needs of the Finnish nuclear sector. In 2017 the hot cells will finally be useable by VTT personnel, so it is expected that significant efforts will be needed to achieve full proficiency of the operators.

As was outlined in the REHOT 2015 project report “Roadmap for the VTT Centre for Nuclear Safety Research Infrastructure” [2.], the commissioning of the new hot cell facilities is not only a technical endeavour, but it requires competence development. The development of remote operation methods and skills is ongoing at VTT, and encompass infrastructure, operational and substance competences. The infrastructure competencies are those skills required to utilize the facilities and equipment correctly, purposefully and effectively. It applies as much to the particular research and testing devices, as to the associated supporting infrastructures. For the research equipment it means the correct operation of the equipment as much as an understanding of the results produced and their accuracy or sources of inaccuracy. The effective utilization of the remote-handling hot cells includes things like skilful manipulator operation, visualization with remotely-operated cameras, and development and utilization of in-cell tools and aids.

When it comes to testing, accreditation is required for the doers of accredited tests. Currently fracture mechanical testing, tensile testing and impact testing of irradiated reactor pressure vessel materials are accredited, but will require renewal in the new testing environment, while other testing methods could also seek accreditation as well.

The HOTLAB 2017 conference has been shown to be an excellent opportunity for learning remote handling approaches from other hot cell users, not to mention developing the network between user facilities. In 2015 the VTT hot cells were presented at the HOTLAB conference to complement the 2014 overview of the whole facility [3.][4.]. In 2017 it will be hosted for the first time in Japan, which will no-doubt prompt a special focus on remote-handling challenges spawned from the Fukushima disaster. In line with SAFIR2014 evaluators encouragement to host more international conferences in Finland, VTT will host the 2018 HOTLAB conference. That requires participation in the management board starting from the HOTLAB 2017 conference. An allocation is included to enable participation in the 2017 conference in Japan.

2.2 Equipment procurement (WP2)

Over the 2016-2018 period this work package is mainly tasked with continuing to execute the procurement process for the hot laboratory equipment that was begun in the REHOT project. The equipment is comprised of “standard” factory-produced devices related to the testing and research to be carried out in the hot cells, microscopy facilities, and nuclear waste management laboratories. The work package involves managing the investment schedule and installation in conjunction with the facility commissioning, executing the competitive bidding process for each item, and assuring that the equipment can be adapted to the hot cell or radiological environment. An important part of
this work package is managing the overall investment schedule. The person months allocated for this work package are intended for executing the procurement, as well as the initial testing and commissioning of the equipment and the training of the users of each new device.

The formation of VTT Ltd in 2015 significantly improved the flexibility in making investments, as the investment aid process can now be utilized. The third VYR funding instrument will be important in supporting these investments, but each investment decision is made by VTT, and the professional organization will support and follow the procurement processes.

In 2017 installation of the hot cells will proceed, and the contract with ITD is expected to conclude. The remaining payments to ITD will comprise the bulk of the investment costs, but a few devices related to getting the new facilities up and running will still be purchased. The current rendition of the investment schedule is found in Appendix B.

The making of new equipment investments is a long process that requires thorough preparation in many stages, including investment preparation, competitive bidding, and installation and inauguration. But each of these phases involves a number of activities, some of which can occur in parallel, but others that can only happen once previous stages have been completed. For example, just to prepare for an investment, an accurate specification must be made, an appropriate selection of potential suppliers must be found, and budgetary cost estimates must be made. To assemble an adequate bid request, iterations of that process might be required based on what is actually available on the market, and on details that may be gleaned from the suppliers themselves. All of these processes are ongoing in parallel for each of the different investments, and clearly form a multi-year process. A Gantt chart of the procurement processes, including those beyond the current project year, are shown in Appendix C.

Once particular suppliers are identified, negotiations and quality assurance evaluations are required. Thus, this work package requires some travel to visit equipment expositions and supplier premises for demonstrations, negotiations, quality assurance visits, factory acceptance tests, etc. When equipment is delivered, installation may also require execution of contracts for special handling, minor facility modifications, and start-up supplies, but in 2017 those will be paid from the Research Infrastructure surcharge, in the “Other” column of the budget.

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2.2.1 Reactor materials mechanical test devices (T2.1)

The procurement of four main devices related to mechanical testing of reactor materials were begun in the 2016 project year. Their procurement will be advanced in the 2017 project year. They include a 450J instrumented impact hammer for standard-sized 10x10 Charpy testing that will be locally shielded in the A-class high bay, an in-cell pre-fatigue device for fracture mechanical test specimens, a computer-numerically-controlled (CNC) machining station for installation in hot cell 3.1, and an in-cell digital measuring microscope for optical quantification of test specimens dimensions before and after mechanical testing. All four devices were given high priority in the investment review conducted in January 2016.

A full-sized pendulum impact tester is used for the determination of impact resistance on plastics, metals and other materials. Testing can be done in accordance with applicable ISO, EN and ASTM standards. Instrumented testing can be utilized to measure the complete force-deflection diagram during impact to obtain detailed material data. The Zwick version of this device includes an integrated specimen tempering and feeding unit for the temperature range -180...600 °C. The procurement of this device is necessary in 2017 to enable integration with the localized shielding that is being developed in Task 3.2.

Pre-fatiguing of specimens for fracture mechanical testing can occupy a single device for a full day, which has proven to be a significant bottle-neck during large RPV material campaigns. As such, a smaller, alternative device to focus on only pre-fatiguing can free up the larger, more versatile mechanical testing devices. After comparison with a hot-cell ready pre-fatigue device dedicated to 10x10 Charpy-type specimens, the trend towards CT-type specimen testing led to the conclusion that an in-cell pre-fatigue device should be capable of pre-fatiguing both Charpy and CT-type specimens. Such a device will be procured in 2017 for integrating into hot cell 1.5 as soon as the cell has been installed.

In order to recover mechanical test specimens from surveillance capsules, a means for machining open the capsule is required. The work-piece must also be fixed securely to the machining table remotely. In 2016 the KUKA
robot was demonstrated for some aspects of this operation, but it was not considered reliable enough or efficient enough for other things, like sawing off of the ends of the capsule. Therefore, a computer-numeric-controlled (CNC) machining station is still foreseen for installation in hot cell 3.1 in the basement of the VTT CNS. The final design and cost estimate and fabrication of the machining device and the work-piece fixing device will be carried out as a part of this task in 2017, culminating in a report describing the design. Finally, the measurement of test specimens e.g., dimensions, notch depth/geometry, lateral expansion, reduction of area, etc. before and after mechanical testing is important for accurately determining the properties of the material being tested. In the remote-operation setting of the hot-cell, the use of hand tools is very clumsy if not impossible. However, very good video microscopes are on the market which can quickly and accurately execute a number of different measurements automatically. In 2017 such a device will be purchased for placing in hot cell 1.4, located next to the mechanical testing cell 1.5.

### 2.2.2 Microscopy devices (T2.2)

This task will be completely removed from the RADLAB project following results of the proposal evaluation. As described in separate reports, analytical microscopy is a fundamental aspect of research on reactor materials. In the CNS such characterization is possible from macro- down to nano-scale, and in the common laboratory facilities the final repository materials research can also now benefit from the top-of-the-line devices. In 2016 the release of the new laboratory enabled the TEM and SEM instruments to be installed in the new facilities. In 2017 the hot cells will be ready for equipping, so it is appropriate to complete the procurement of the cell metallography devices. This includes remotely operable devices for precision cutting of samples, mounting of samples, grinding and then polishing, as well as electro-polishing and etching. While a few of the new devices exceed the 10 k€ threshold for investments (e.g. hot-mount press equipment, grinding and polishing devices) some are already available from existing equipment and others can be purchases as normal laboratory supplies rather than as an investment.

In 2016 a Zeiss Crossbeam FEG equipped scanning transmission electron microscope and analytical microscopy detectors were purchased according to plan. As was described in the report about the microscope selection, the instrument was specified to accommodate the addition of a focused ion beam (FIB) device for site-specific extraction of micro-specimens. The intention was to complete the procurement of the device in 2016 by purchasing the FIB accessory. Because the device was prioritized lower than almost all other proposed investments, it has not yet been purchased though. Instead, a research infrastructure proposal has been made to the Academy of Finland for purchasing the FIB with 100% Academy of Finland financing. However, other research groups at VTT have also expressed an interest in the device, so as an alternative plan in case the Academy of Finland funding is not obtained, an investment allocation is proposed to pay for half of the costs from VTT’s research investment budget, with the other half of the costs proposed for inclusion in the investment aid budget of RADINFRA.

### 2.2.3 Nuclear Waste Management laboratory devices (T2.3)

In accordance with the SAFIR evaluations of the proposal, no SAFIR funds will be used for this task. Since about half of the laboratory wing of the CNS is dedicated to nuclear waste management research laboratories, two new investments are proposed for purchase using the investment aid instrument: a replacement liquid scintillation counter for radiochemistry, and a materials testing device for bentonite research.

A liquid scintillation counter (LSC) is used to measure the beta and alpha particle activity of radioactive materials. Active material is mixed with liquid scintillator, which reacts to radioactive decay and emits visible photon emission which are recorded and counted. The existing LSC equipment is over 20 years old and maintenance and spare parts are difficult to obtain. The new LCS equipment would be used to analyse C-14 in the nuclear power plant decommissioning waste dissolution studies and to analyse radium in nuclear waste management studies. The new LCS equipment will update the activity measurement capabilities of the CNS and ensure the future usability of LCS equipment in CNS. In the spirit of interdisciplinary utilization of the common laboratory facilities, the LSC equipment can be used for Nb-measurements of pressure vessel dosimetry and national and international commissions for neutron dosimetry of nuclear power plants materials as well.

Materials testing devices for bentonite research (including a load frame, a pressure-volume controller for pore pressure and an operating system) are utilized in experiments on the mechanical properties of bentonite. Together with VTT’s (already existing) high pressure triaxial cell, the devices allow building a fully controlled testing environment for stiff soil materials, such as highly compacted bentonite and rock. The control of the stress path in the experiments is especially important for the application of bentonite clay as the buffer material in the spent nuclear fuel disposal system, since the mechanical behaviour of bentonite depends on its current pressure state, which may
vary in the disposal system. Therefore, the characterization of the buffer material in the disposal conditions requires the use of a fully-controlled testing environment. The devices also allow to further develop the method of pore water squeezing that is used in the pore structure and chemical analysis of bentonite.

2.3 Research equipment (WP3)

This work package is mainly tasked with the development and construction of those research devices that are not readily available on the market, but rather, require custom design. Such devices are designed with the experts involved in utilizing the equipment for producing research results, and are then made by in-house assembly of parts bought from component suppliers, or fabricated in-house or by outside shops. In 2017 incidental costs of materials, supplies and external services beyond those included in investments, will be paid from the Research Infrastructure surcharge, in the “Other” column of the budget. A Gantt chart of the activities, including those beyond the current project year, are shown in Appendix C.

This work package also includes managing the moving and installation plan for all research equipment in the new facilities, whether it is existing equipment, newly purchased equipment, or devices that have been custom fabricated earlier or during the course of the RADLAB project. It is expected that functional adaptation work will be required for many pieces of equipment, as well as purchase and installation of incidental systems enabling them to be used in the environment. Thus, this work package also includes the nuclearization of equipment that VTT will be tasked with integrating into the hot cells.

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<th>Partners in WP3</th>
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2.3.1 Hot autoclave fabrication (T3.1)

The hot autoclaves enable safe mechanical testing of radioactive materials in well-controlled, simulated power plant water environments. The main use for such facilities currently is in assessing the susceptibility of stainless steel internals component materials to irradiation assisted stress corrosion cracking (IASCC), generally as a function of material irradiation condition, loading scenario, and water chemistry. For that reason, this task is linked to the Euratom SOTERIA project which started in 2015. That project studies IASCC susceptibility, so the intention is that the infrastructure for conducting such tests will be constructed in the RADLAB 2016 and 2017 projects, while the device functionality will be demonstrated in the SOTERIA project (refer to Figure 3).

The materials testing autoclaves and water circuit are to be located in a dedicated room of the basement of the CNS. The floor plan/placement of the hot autoclaves and water circuits for the new facility was first done in 2015 [8.] and the plans were updated in 2016. In 2016, the needs for upgrades for old hot autoclaves (2) and water circuit (1) were identified and offers requested from suppliers. In the last pressure vessel/autoclave inspection a crack indication was found in one of the two old autoclaves. Due to the contamination and general condition of that autoclave, a decision was made to replace it instead of renewal. Components for the renovation of the other autoclave were ordered and will be delivered by the end of 2016. An offer for a new autoclave was requested and received. Due to budgeting issues, upgrades for the water circuit and the fully new autoclave will be ordered in 2017. The water circuit and the autoclaves will be installed in the CNS during 2017. However, due to the relatively low priority given this task in the proposal evaluation, the final installation and description of said installation will not be carried out as a part of this project.

2.3.2 Nuclearization and in-cell devices (T3.2)

With the installation of the new hot cells taking place in 2017, the nuclearization and installation of in-cell devices and locally shielded devices takes high priority. In conjunction with the on-site assembly of the hot cells, the electron-beam welder (EBW) and the electric discharge machine (EDM) will be installed directly into their respective shielding cells, aided by ITD’s personnel. Likewise, small table-top devices must be adapted for their new in-cell environment more completely when the hot cells are assembled on site.
In 2016 some needed modifications to the in-cell EBW were identified and the decision was made to purchase the modification plan and execution from the EBW manufacturer. Thus, in 2017 the modification will be made in time for installation of the EBW into the new hot cell 1.2.

The need for a remotely-operated door on one side of the EDM device to facilitate handling of heavy work pieces was considered in 2016. Ultimately it was abandoned during the further detailed hot cell design process though, in favour of developing a front-loading device that can be more readily viewed and operated at the hot cell work station. This front-loading device will have to be designed and built once the EDM has been installed into the hot cell. Also, the EDM water circuit that was developed in the earlier REHOT and RADLAB projects will have to be evolved to the final, more compact and effective design once the EDM has been installed in the hot cell. This is expected to take place in the latter part of 2017.

There are a number of small devices and aids required for remote operation when using the hot cells. Many of them will require modifications, but the materials and supplies for them can be purchased with the proceeds from the research infrastructure surcharge. Such devices include the metallography equipment of Task 2.2, as well as other devices such as the microhardness tester, optical dimensioning microscope and mechanical testing aids.

Finally, while the main equipment for testing and characterization of radioactive structural materials is located in the hot cells constructed by ITD, non-active materials, contaminated materials, and materials of low levels of radioactivity can be handled more quickly and easily in locally shielded devices. This can relieve some of the work-load of the hot cells themselves, and increase the overall capacity of the facilities. The intention is to employ local shielding initially for a full-size 450 J instrumented impact hammer (replacing the old, outdated device that will be decommissioned with the OK3 facilities) and a universal mechanical testing device. The conceptual design of the local shielding for those devices took place in 2016, and the engineering design and fabrication will take place in 2017. The shielded devices will be up and running in time for radiological commissioning of the whole facility at the end of 2017. Local shielding will include, at a minimum, a containment enclosure attached to the ventilation system so that any possible contamination is isolated. For lightly gamma irradiating materials, some local lead shielding will be employed as needed. Handling will be via glove-ports and utilizing hand tongs to offer radiation protection.

2.3.3 Nuclear Waste Management auxiliary items (T3.3).

In accordance with the SAFIR evaluations of the proposal, no SAFIR funds will be used for this task. Only KYT funds will be used. In addition to the device procurements outlined in Task 2.3, some auxiliary items must be procured for improving existing devices that have been moved into the new nuclear waste management laboratories of the CNS. These included oxygen analyser and sensors for the anaerobic glove boxes, as well as a portable temperature sensor system with data acquisition. This task procures, installs and commissions those devices.

There are 7 glove boxes in the CNS that are used to conduct experiments under controlled atmosphere. At the moment glove boxes are anaerobic in argon atmosphere and are used to simulate the final disposal conditions of spent nuclear fuel. Studies simulating final disposal of nuclear waste need confirmation of anoxic conditions in the studied systems. In order to maintain the anoxic atmosphere, oxygen monitoring in the glove boxes is essential. Oxygen levels in the glove boxes are recorded with Orbisphere analysers, which enable low enough values ( < 0.01 ppb) to be measured in the solutions used in experiments inside an anaerobic glove box. Each glove-box also needs an oxygen sensor in the purification circulation to indicate when the gas purifier of the glove box needs to be serviced, so that the required low oxygen content of the gas atmosphere is maintained at the level required for the studies inside the glove box. The sensors need to occasionally be replaced.

Temperature also has a great affect in chemical reactions and physical phenomena which are important in research related nuclear waste disposal. Many observable changes in experiments can be due to changes in temperature. If temperature changes are not recorded during the experiments, observations may lead to incorrect interpretation of results. Certain tests are also regulated by standards that require a specific temperature during the test, so monitoring and recording the temperature is essential to successfully fulfill standard criteria. Therefore, a new temperature system with changeable thermocouple probes will be purchased to enable measuring temperature from different kind of experiments in the laboratory. A new system will also increase the accuracy of measurements, and enable more complete information from on-going and future experiments for interpretation of results.

2.4 Supporting facilities (WP4)

The VTT CNS requires a number of self-built supporting facilities. Upon conducting the hot cell Conceptual Design and Detailed Cost Assessment in 2013, 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. As started in 2016, this work package will continue
in 2017 to construct and install the equipment for three main areas: laboratory radioactive waste handling, radioactive research material logistics, and orderly storage of radioactive specimens. The hardware for these systems are mainly located in the basement of the CNS. A Gantt chart of the activities, including those beyond the current project year, are shown in Appendix C.

In each area a conceptual design has been completed, and the facility is being engineered for manufacture or assembly from off-the-shelf components. As with WP3, the systems are designed with the personnel involved in utilizing the equipment for supporting the testing and research activities, and are then realized by in-house assembly of devices and parts bought from suppliers, and/or made by in-house or by outside shops on contract. In 2017 incidental cost of materials, supplies and external services beyond those included in investments, will be paid from the Research Infrastructure surcharge, in the “Other” column of the budget.

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### 2.4.1 Waste handling system (T4.1)

The design requirements and some options for the waste handling system were elaborated in a report produced in the REHOT 2015 project on a contract with Platom Oy [9.] A separate room is available for wet waste handling installations, including the EDM water loop and a water evaporation/re-condensation system. A separate room is also available for the dry waste handling installation, including waste sorting, drying and packing equipment. In 2016 the main focus was on the dry-waste handling, and effort went to ensuring that the building fixtures, epoxy floor and final as-built dimensions were suitable for the purpose. The light-duty bridge crane was procured, and the concrete shielding blocks were sourced. However, for the wet waste handling the intended contract for producing the glove box did not progress as quickly as originally foreseen, since the need to carry out a full tender process for outsourcing the work proved challenging without first having a clear design proposal. The decision to delay licensing of that part of the A-class facilities also reduced the schedule pressure. However, in this 2017 project proposal there is a greater emphasis on including the nuclear waste management experts in the project, and it is expected that this will bring much needed radiochemistry expertise for developing sustainable methods for treating and handling the liquid waste produced in the radiological laboratories. Therefore, the focus of this task in 2017 is on developing the liquid waste handling methodology and realizing the required facilities. This involves both the solid particle separation technology that is primarily for the EDM water circuit, and the neutralization, concentration and decontamination of activated liquids.

### 2.4.2 Specimen storage system (T4.2)

Means are required for safe and orderly interim storage of radioactive test materials. The location for a primary storage facility has been specified in the new CNS to take advantage of natural shielding offered by the surrounding granite. A comprehensive storage system consists of both a shielded storage device and the means for safely and orderly picking and placing of the radioactive materials, as well as an inventory database for maintaining important specimen information and tracking its location in the facility when it is taken out of the storage facility.

In 2016 the conceptual design of the physical storage device was completed and its engineering design and fabrication will be realized in 2017.

An important aspect of the move from the old OK3 facilities is the orderly transfer to the new facilities of any radioactive specimens that cannot otherwise be returned to their origin by that time. To prepare for that, a new database is being created in which the existing materials inventory can be recorded, and which can then also form the basis for the inventory database system in the new facilities. In 2016 a contract was launched with Ambientia Oy for realizing the first specimen database system, which includes functionality for tracking specimens throughout the radiological facilities as well as pinpointing storage locations in the central storage device. In 2017 this database will be populated with the specifics of the materials that will be stored in the new facilities.
2.5 VTT CNS (WP5)

During the 2016 to 2018 period the VTT CNS will be commissioned, and taken into use for providing research and testing services on radioactive materials domestically and internationally. This involves coordination and planning for competence development, continued assessment of research growth areas, discovery of funding and income possibilities, and building the market for contract work (domestic and international). Because the RADLAB project is also coordinating the SAFIR RG6 meetings, a small allocation is included for covering the costs associated with hosting meetings, enabling some refreshments to be provided.

In 2017 this task especially reflects the fact that half of the new laboratory space is dedicated to nuclear waste management research. As such, two main tasks are identified: development of the ICP-MS competencies for broader utilization of this technology, and the commissioning of the A-class facilities themselves.

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2.5.1 HR ICP-MS development and training (T5.1)

In accordance with the SAFIR evaluations of the proposal, very little SAFIR funds will be allocated for this task. Instead, most of the KYT funding will be allocated to this. The licensing application of the B- and C-class radiological facilities proceeded as planned in 2016, with both the safeguards and the radiological laboratory licence applications being submitted to STUK. In 2017 the utilization of the new facilities will be ramped up, as final devices are moved in along with the radioactive materials that they are testing. The most important of these is the HR ICP-MS (High Resolution Inductively Coupled Plasma Mass Spectrometer).

The HR ICP-MS is a versatile instrument for quantitative and qualitative analyses of most elements (excluding H, C, N, O, F, Cl and noble gases). The high resolution of the equipment also enables the analysis of isotopic ratios. The concentration range covers the analyses from the mg/L to sub pg/L. The instrument is ideal for liquid samples with trace level concentrations like; pure chemicals, semiconductor, geological and material science samples.

In CNS the HR ICP-MS equipment is placed in a clean room, which offers many advantages concerning analysis of low concentrations. Background in the metal-free cleanroom environment is significantly lower than in a normal laboratory. ICP-MS analyses also serve many analysis and research purposes in the nuclear field. For example, it has been used to measure low concentrations of uranium isotopes in solubility studies, boron isotopic ratios in process waters, and low levels of Si, Al, Fe and Mg in bentonite waters. It is also a great tool for determining elements at trace levels in different reactor materials.

The new facilities in the CNS offer new opportunities to develop elemental and isotopic analyses carried out with HR ICP-MS. In addition to Argon as a plasma gas, the new clean room also has connections to mixed methane/Ar and N₂ gases. The addition of a small amount of methane to Ar stabilizes the plasma behaviour and improves the sensitivity of the instrument. The N₂ gas was implemented for an additional APEX sample introduction device, which decreases the time needed for sample rinse-out and increases the sensitivity and by removing the solvent from the sample solution. Both of these systems, the mixed methane/Ar gas and APEX sample introduction system are new to the ICP-MS users. Their deployment and the testing of the effects for analysis sensitivity to different elements will be conducted in this task.

A round robin test is an inter-laboratory test (analysis or measurements) performed independently several times. These kind of test programs usually involve several (10 to 30) laboratories with one or a few techniques to analyse a list of elements. This kind of test can be considered as worldwide bench-marking test and as a validation of methods used in different laboratories. VTT had traditionally taken part in such round robin tests regularly, but the new facilities and research environment definitely require participation in new round robins. This project enables the participation and test analyses conducted in a round robin.

To utilize the HR ICP-MS equipment to its full potential, further training of the HR ICP-MS experts is also needed. 3 of the 4 operators participated in “the basics of HR ICP-MS” course offered with the instrument installation 8 years ago. To enhance capabilities of the instrument, the operators should be further trained to build up their expertise. There are several good courses arranged by either the instrument provider ThermoScientific or by the international ICP-MS communities, to learn more about the different applications. Participation of 1 to 2 operators to the selected available courses will be accomplished in this project.
This project will enhance both the capability of the ICP instrument and the know-how of the operators of it. The better sensitivity of ICP-MS enables the detection of even lower elemental concentrations. The enhancement of know-how and quality control of the analysis will enable the technology to serve the nuclear field in Finland with better expertise.

2.5.2 Commissioning of A-class rooms, project management (T5.2)

The commissioning of the VTT CNS is not only a technical undertaking, but it also includes the organizational and business aspects of launching this new flagship infrastructure. While the installation and functionality testing of the devices in their new environment is important, of greatest importance is the successful radiological commissioning of the facility and its equipment. The new devices and lab spaces cannot be used for testing and research on radioactive substances until they have been approved by the Radiation and Nuclear Safety Authority STUK. Following discussions with STUK, the decision was made to apply for a single new radiological permit for the new facilities, leaving the existing permits in place for the old facilities during their decommissioning process. The new permit was applied for in 2016 to initially cover those activities which are the most straightforward to get up and running correctly and safely, encompassing in the B- and C-facilities. This will then be expanded in 2017 to the A-class facilities as the hot-cells are installed and can be commissioned.

While the commissioning is the main technical goal of this task, the overall management of the project is also executed as a part of this task. This comprises not only the RADLAB project management for SAFIR, but also the coordination of this Centre for Nuclear Safety infrastructure renewal effort involving the equipment procurement portfolio of RADINFRA and the facility itself of RADCNS.

To expand the utilization potential of the new facility, in 2016 a big effort was made to updated the business offerings, to develop marketing materials like a video, lobby display and the VTT webpages, and to raise the awareness of the new infrastructure by touring numerous different national and international groups through the new facility.

In 2017 the technical commissioning will be complemented by updating the quality assurance processes and accreditation of specific tests executed in the new facilities, as well as producing a significant portion of the operational guides for the new laboratory. In accordance with the SAFIR evaluations of the proposal, the SAFIR funds will be used only for the project management aspects of this task.
3. Deliverables 2017

Since the RADLAB project is focused on infrastructure realization, the deliverables are mainly functional devices. Therefore, based on the experience in 2016, the progress follow-up in 2017 will mainly focus on milestones and summary reports rather than on scientific reports and publications. Scientific reporting is carried out in research projects utilizing the infrastructure, while the RADLAB project is focused on producing the tools themselves.

<table>
<thead>
<tr>
<th>Deliverable number</th>
<th>Deliverable name (and description if necessary)</th>
<th>Indicative person months</th>
<th>Deadline date</th>
</tr>
</thead>
<tbody>
<tr>
<td>M1.1.1</td>
<td>Installation and Pre-commissioning of Group 1A - 1.3 / 1.6 / 1.7</td>
<td>4</td>
<td>Mar. 2017</td>
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<tr>
<td>M1.1.2</td>
<td>Installation and Pre-commissioning of Group 1B - 1.1 / 1.2 &amp; Cell Group 3 - 3.</td>
<td>4</td>
<td>May 2017</td>
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<tr>
<td>M1.1.3</td>
<td>Installation and Pre-commissioning of Group 1C - 1.4 / 1.5 &amp; Cell Group 2 - 2.</td>
<td>3</td>
<td>Jul. 2017</td>
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<tr>
<td>D1.1.1</td>
<td>User’s Guide for as-manufactured hot-cells</td>
<td>1</td>
<td>Oct. 2017</td>
</tr>
<tr>
<td>M1.2.1</td>
<td>Completion of training of several hot cell users by ITD</td>
<td>4</td>
<td>Dec. 2017</td>
</tr>
<tr>
<td>D2.1.1</td>
<td>Design report of custom in-cell CNS machining station</td>
<td>0,5</td>
<td>Jun. 2017</td>
</tr>
<tr>
<td>M2.1.1</td>
<td>Delivery of fracture mechanic specimen pre-fatigue device for hot cell installation</td>
<td>0,5</td>
<td>Jun. 2017</td>
</tr>
<tr>
<td>M2.1.2</td>
<td>Delivery of optical dimensioning microscope</td>
<td>0,5</td>
<td>Jul. 2017</td>
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<tr>
<td>M2.1.3</td>
<td>Delivery of large Charpy impact testing device</td>
<td>0,5</td>
<td>Nov. 2017</td>
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<td>D2.2.1</td>
<td>Functionality test of in-cell metallography specimen preparation line</td>
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</tr>
<tr>
<td>M3.1.1</td>
<td>Acquisition of key components of “hot” autoclave facilities</td>
<td>2</td>
<td>Oct. 2017</td>
</tr>
<tr>
<td>D3.2.1</td>
<td>Engineering design of locally shielded equipment installations</td>
<td>3</td>
<td>Nov. 2017</td>
</tr>
<tr>
<td>M3.2.1</td>
<td>In-cell installation of nuclearized EBW device</td>
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<td>May 2017</td>
</tr>
<tr>
<td>M3.2.2</td>
<td>In-cell installation of EDM</td>
<td>3</td>
<td>Jul. 2017</td>
</tr>
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<td>D4.1.1</td>
<td>Descriptive report of waste handling installation</td>
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<td>M4.2.1</td>
<td>Engineering and fabrication contract for specimen storage system</td>
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<tr>
<td>D5.1.1</td>
<td>Report of round robin test of HR-ICP-MS in new clean room setting</td>
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<td>Nov. 2017</td>
</tr>
<tr>
<td>Deliverable number</td>
<td>Deliverable name (and description if necessary)</td>
<td>Indicative person months</td>
<td>Deadline date</td>
</tr>
<tr>
<td>--------------------</td>
<td>---------------------------------------------------------------------------------------------------------------</td>
<td>--------------------------</td>
<td>---------------</td>
</tr>
<tr>
<td>D5.1.2</td>
<td>Training certificates of two additional operators for HR-ICP-MS</td>
<td>2</td>
<td>Dec. 2017</td>
</tr>
<tr>
<td>M5.2.1</td>
<td>Submission of licensing papers for A-class facilities</td>
<td>Z.5 3</td>
<td>Dec. 2017</td>
</tr>
<tr>
<td><strong>KYT-only Milestone</strong></td>
<td><strong>Milestone name</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>M3.3.1</td>
<td>Commissioning of oxygen sensors</td>
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<td>Apr. 2017</td>
</tr>
<tr>
<td>M3.3.2</td>
<td>Commissioning of temperature loggers</td>
<td>0.5</td>
<td>May. 2017</td>
</tr>
<tr>
<td>M2.3.1</td>
<td>Delivery of liquid scintillation counter</td>
<td>0.5</td>
<td>Oct. 2017</td>
</tr>
<tr>
<td>M2.3.2</td>
<td>Delivery of bentonite mechanical test station</td>
<td>0.5</td>
<td>Nov. 2017</td>
</tr>
</tbody>
</table>

*Note: Physical devices are generally joint deliverables with the RADINFRA project*
4. Project organisation

The RADLAB project is an integral part of the overall infrastructure renewal process surrounding the VTT CNS, as was illustrated in Figure 1. This process is executed by the organization shown in Figure 6. The RADLAB project itself is executed by VTT, and is managed by an IPMA Level C certified project manager, Dr. Wade Karlsen. However, the project operates under the auspices of the VTT CNS Steering Group that includes, among others, Erja Turunen (EVP Smart industry and energy systems, IND), Satu Helynen (Vice President of Nuclear Safety), and Tanja Huopanen (CFO). The Technical Lead for the hot cell construction work package is Seppo Tähtinen. The main researchers, their organisation, the tasks that they will be contributing to, and their estimated person months in 2016 are shown in the table on the next page.
<table>
<thead>
<tr>
<th>Name</th>
<th>Title</th>
<th>Organisation</th>
<th>Participates in tasks</th>
<th>Estimated person months (2015)</th>
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<tbody>
<tr>
<td>Seppo Tähtinen, MScTech</td>
<td>Senior Scientist</td>
<td>VTT</td>
<td>all</td>
<td>6.8</td>
</tr>
<tr>
<td>Ilkka Palosuo, MSc</td>
<td>Research Engineer</td>
<td>VTT</td>
<td>T1.2, T2.1, T3.2</td>
<td>5</td>
</tr>
<tr>
<td>Mika Jokipi, TechEng</td>
<td>Research Engineer</td>
<td>VTT</td>
<td>WP1, T3.2, T4.2</td>
<td>5</td>
</tr>
<tr>
<td>Wade Karlsen, Ph. D.</td>
<td>Research Team Leader</td>
<td>VTT</td>
<td>all</td>
<td>5</td>
</tr>
<tr>
<td>Arto Kukkonen, Tech</td>
<td>Senior Research Technician</td>
<td>VTT</td>
<td>all</td>
<td>3</td>
</tr>
<tr>
<td>Jarmo Siivinen, TechEng</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T4.1</td>
<td>3</td>
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<tr>
<td>Kimmo Rämö, Tech</td>
<td>Research Technician</td>
<td>VTT</td>
<td>T3.2, WP4</td>
<td>3</td>
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<tr>
<td>Pekka Moilanen, DrTech</td>
<td>Senior Scientist</td>
<td>VTT</td>
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<tr>
<td>Tuomo Lyytikäinen, TechEng</td>
<td>Research Engineer</td>
<td>VTT</td>
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<td>2</td>
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<tr>
<td>Jari Lydman, MSc</td>
<td>Research Engineer</td>
<td>VTT</td>
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<td>1</td>
</tr>
<tr>
<td>Marko Paasila, Tech</td>
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<td>VTT</td>
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<td>1</td>
</tr>
<tr>
<td>Tiina Lavonen, MSc</td>
<td>Research Scientist</td>
<td>VTT</td>
<td>T2.3, T3.3, T5.1</td>
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</tr>
<tr>
<td>Aki Toivonen, PhD</td>
<td>Senior Scientist</td>
<td>VTT</td>
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</tr>
<tr>
<td>Emmi Myllykylä, MSc</td>
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<td>Research Technician</td>
<td>VTT</td>
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<tr>
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</tr>
<tr>
<td>Kirsti Helosuo, TechEng</td>
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<td>VTT</td>
<td>T3.3</td>
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<tr>
<td>Pasi Väisänen, Tech</td>
<td>Engineer</td>
<td>VTT</td>
<td>T3.1</td>
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<tr>
<td>Tiina Heikola, MSc</td>
<td>Research Scientist</td>
<td>VTT</td>
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<td>Veli-Matti Pulkkanen, MSc</td>
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<td>VTT</td>
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<tr>
<td>Juha-Matti Autio, MScTech</td>
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<td>T2.2</td>
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<tr>
<td>Matias Ahonen, DrTech</td>
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<tr>
<td>Mykola Ivanchenko, DrTech</td>
<td>Research Scientist</td>
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<tr>
<td>Petteri Lappalainen, MScTech</td>
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<td>VTT</td>
<td>T1.2, WP4</td>
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<tr>
<td>Tommi Kekki, MScTech</td>
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<td>VTT</td>
<td>T1.1, T2.3, T5.2</td>
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</tr>
<tr>
<td>Ulla Ehrnstén, MScTech</td>
<td>Research Technician</td>
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<td>T2.2</td>
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<tr>
<td>Ulla Vuorinen, MSc</td>
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<tr>
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<td>Senior Scientist</td>
<td>VTT</td>
<td>T2.2</td>
<td>0</td>
</tr>
</tbody>
</table>

**SUM** 46.8
5. Risk management

Because the RADLAB project is a part of the overall broader infrastructural renewal program being undertaken at VTT, it is subject to many of the same risks as the overall infrastructural renewal, and also has some internal risks of its own.

The principle risks specific to the RADLAB project are described in the table shown below. The table does not include risks that have a combined risk level \( (= \text{Likelihood} \times \text{Severity}) \) of 2 or less, nor standard risks such as key personnel leaving VTT, nor unpredictable events such as strategic direction changes by key stakeholders, etc.

<table>
<thead>
<tr>
<th>Risk</th>
<th>Likelihood (1-4)</th>
<th>Severity (1-4)</th>
<th>Risk level (L x S)</th>
<th>Action</th>
</tr>
</thead>
<tbody>
<tr>
<td>Insufficient project funds for completion of required tasks</td>
<td>3</td>
<td>2</td>
<td>6</td>
<td>Seek additional support from VTT or accept delay in some lower priority items</td>
</tr>
<tr>
<td>Insufficient VTT personnel resources for commissioning phase</td>
<td>3</td>
<td>2</td>
<td>6</td>
<td>Prioritize certain commissioning activities; Add people from within VTT</td>
</tr>
<tr>
<td>Lack of personnel willing or able to work in new hot cell facility</td>
<td>2</td>
<td>3</td>
<td>6</td>
<td>Recruit from abroad (may not be competitive); hire and train new people</td>
</tr>
<tr>
<td>Financial demands exceed VTT capacity to make key investments</td>
<td>2</td>
<td>2</td>
<td>4</td>
<td>Reduce scope of infrastructural renewal or significantly delay some procurements</td>
</tr>
<tr>
<td>Insufficient VTT personnel resources for self-building of facilities</td>
<td>2</td>
<td>2</td>
<td>4</td>
<td>Outsource a greater portion of the design and fabrication work; accept slower progress</td>
</tr>
<tr>
<td>Unforeseen technical problems in commissioning particular facilities</td>
<td>2</td>
<td>2</td>
<td>4</td>
<td>Seek resolution via dialogue with authorities, and implement required technical changes</td>
</tr>
</tbody>
</table>
References


Liite_2_APPENDIX A

DRAFT PROJECT SCHEDULE FOR HOT CELL REALIZATION WITH ITD.
Liite_2_APPENDIX B

PROCUREMENT SCHEDULE FOR INFRASTRUCTURE RENEWAL
SAFIR2018 RADLAB Appendix B: Ydinturvallisuustalo: Kuumakammiolaboratorio-investoinnit

1027
322,3

5427,475
1703,453

4732
1485

Kustannusarvion värikoodit: varmistunut/toteutunut hinta

RADINFRA
2015-2020

Tarve SAFIR 2018
Tarve KYT 2018

2740

-1118

2740

-2740

2740

-2740 KattoSAFIR

13 700

860 -350,9

860

-860

860

-860 Katto KYT

0
0

TarvSAFIR
Tarve KYT

4 300
12 808
4 020

1622
509,1

0
0

arvio perustuen budjettitarjoukseen

VTT

1992

860 625,1

RADINFRA

2740

1166

VTT

3714

860

RADINFRA

2740

0

2020

VTT

0

Katto KYT 2018

2019

RADINFRA

VTT laina

Katto SAFIR 2018

VYR jako SAFIR/KYT

VTT

RADINFRA

2018

VTT laina

2017

RADINFRA

2016

RADINFRA

REHOT VTT

REHOT VYR

2015

Yhteensä
2015-2020

karkea arvio

31.1.2017 PÄIVITETTY TAULUKKO

Radioactive material database system
Tekninen tukitoiminta
Muuta (ei varsinaisia investointeja)
Jätekammiota vastaavat varustukset
alakertaan
EB-hitsauslaitteen modifiointi
Orbisphere oxygen analyser
Bentonite MTS device
Liquid scintillation counter
Laitteiden nuklearisointi

VTT

RADINFRA

VTT

0

450

0

449

230

0 1 349

81

7 131 1 430 6 217

450 2 131

845

0

450

0

449

0
774
395
1 350
230

575

6 109

4 716

5

774

1 500

1 350

11 630

6 914

484

0
484

267 tilattu, toimittu 11/2016 ITD:lle

tilattu, toimitetaan 11/2017
siirtyy vuodelle 2017
ei hankita ilman selvä tarve

5

selvitystyöt käynnissä

80
26

700

hankinta käynnissä
hankinta käynnissä
26 hankittu; toimitettu 10/2016

39

39 tilattu, toimitettu 9/2016

korvattu pumppukäyryllä

10

ITD:n toimituksissa

150
61

41
41

Toimitettu. Kammioon 8/2017

10

22
40
244
198
0
78
500
10

78
500

7 hankinta käynnissä

selvitys käynnissä
selvitys käynnissä; siirrettiin v. 2018
suunnittelu tehty, ei valmistusta vielä

150
500
20

26

ei vielä hankittu
ei vielä hankittu
ei vielä hankittu
5 Yksi stereomikroskooppi hankittu 9/16
ei vielä hankittu
tarjouskilpailu käynnissä
selvitystyöt käynnissä

20
70
50

60
150
78

6 684 asennus käynnissä

267

7
81

45

toistaiseksi siirretty VTT:n maksettavaksi

302 tilattu, toimittu 11/2016 ITD:lle

30
81

39

774 toimitettu ja asennettu

302

30

150
500
20

Ei hankita

siirtyy mahd. vuodelle 2018
siirtyy mahd. vuodelle 2018

23

100
10

8 463 16 828

230 2 000 Budjetoidut
Toteutuneet ja
230 3 705 arvioidut

Ei hankita VYR:n avulla

30

26

69
20

774
500

60
24
0
15
33

81

80

0

60
24
0
10
33
45

7

10 204

12

350
350
150
60
20

35
265
29
60
0

267

20 763

85
65

85
65
302

18 000

VTT

RADINFRA

845

Toistaiseksi
toteutuneet
RADINFRA
2015-2016

REHOT VYR

VTT

450 2 131

Toistaiseksi
Budjetoitu toteutuneet
yhteensä yhteensä
2015-2020 2015-2016

UUSI VYR/KYT

RADINFRA

7 131 1 430 6 217

VTT

RADINFRA

2020

81

VTT

RADINFRA

2019

0 1 349

SEM pyyhkäisyelektronimikroskooppi
SEM-FIB lisälaite

Kuljetusastian kärry, jolla kuljetusastiaa
liikutetaan sulkutilan ja kellarin välissä
Jätetynnyrien kuljetusta varten
sähkötoiminen trukki tai muu
kuljetusväline apuvälineineen
Sisäisten astioiden kuljetusta varten
väline
Kattonosturi
kuivajätetilaan/näytevarastoon
varustettuna tarttujalla
Dosimetria
Kontaminaatio
Kiinteä säteilyvalvonta
Nuklidimittausjärjestelmä
Jätteiden kuivaus- ja puristuslaitteet
Näytevarasto
Ydinainevarastointijärjestelmä

2018

230

robotti-ovi lankasahalle

TEM läpivalaisuelektronimikroskooppi
Kuumakammiot
tarjottu kokonaisuus
Avauskammio
Kiinnitys/ohjaus/leikkaus
Jyrsinkone
Aineekoetuskammioiden päälaitteet
Uusi iskuvasara
Uusi iskuvasara
Uusi Zwick sähkömekaaninen
Esiväsytyslaite
korkealämpötilauuni
Pienlaitteet kammioissa
Anturit (LVDT, CMOD) (6 * n. 10 ke)
Kamerat (2 * 8 * n. 1,5 ke)
Lämpölevyt (3 * n. 0,1 ke)
Mikroskoopit
Timanttileikkuri
Mikroskopiakammio
mittamikroskooppi
Optinen mikroskooppi
Glove box kammion
metallografiavalmiudet
Muoviin valu- ja syövytyslaitteet
Puoliautomaattihionta- ja
kiillotuslaitteet
Kovuusmittaus
Ympäristövaikutteinen testaus
Uudet pienet autoklaavit
Vesipiirin korjaus
Vanhan pienen autoklaavin korjaus
Näytteiden kuljetus
Kuljetusastia, pieni
Kuljetusastia, iso
Kuljetusastia SEM

2017

VTT

2016

RADINFRA

REHOT VTT

Laitehankintakustannukset
Budjetoidut kustannukset
Toteutuneet ja arvioidut
kustannukset

REHOT VYR

2015

150

150

150

149

599

300

300

300

300

1 200

60
30
98
30

tilattu, toimitus viivästyy toimittajalta
49
150
72

49 tilattu; toimitettu 6/2016
150 tilattu; asennettu 8/2016
72 tilattu; asennettu 8/2016

ollut listalla, mutta ei aiemmin budjetoitu
Siirretty vuodelle 2017
Hankinta käynnissä
Sisälletty VTT:n omaan kalusteesiin
69
20

69 Valmistuu 1/2017; total n.110 k€
20 95% toimetettu/asennettu; lasku 2017

Ei vielä toteutettu
SG6 hyväksyi 6/16
Uusi ehdotus in RADLAB 2017
Uusi ehdotus in RADLAB 2017
Uusi ehdotus in RADLAB 2017
Ei vielä toteutettu


Liite_2_APPENDIX C
GANTT CHARTS FOR WP2, WP3 AND WP4
## Work Package 2 Investments

### TASK 2.1 Mechanical Test Devices for Integration

<table>
<thead>
<tr>
<th>Year</th>
<th>Jan</th>
<th>Feb</th>
<th>Mar</th>
<th>Apr</th>
<th>May</th>
<th>Jun</th>
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<th>Sep</th>
<th>Oct</th>
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<tbody>
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<td>2016</td>
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</tr>
</tbody>
</table>

- **Supplier search (tech requirements, budgetary offers)**: Supplier search (tech requirements, budgetary offers)
- **Selection of CNC machining station for hot cell installation**: Selection of CNC machining station for hot cell installation
- **Investment proposal (VTT template)**: Investment proposal (VTT template)
- **Competitive bid process (notice, evaluation, decision)**: Competitive bid process (notice, evaluation, decision)
- **Procurement Decision and placing of order**: Procurement Decision and placing of order
- **Manufacturing by supplier**: Manufacturing by supplier
- **Delivery by supplier**: Delivery by supplier
- **Nuclearization and Installation by VTT**: Nuclearization and Installation by VTT

### M2.2: Nuclearization and Installation by VTT

- **Investment proposal (VTT template)**: Investment proposal (VTT template)
- **Competitive bid process (notice, evaluation, decision)**: Competitive bid process (notice, evaluation, decision)
- **Procurement Decision and placing of order**: Procurement Decision and placing of order
- **Manufacturing by supplier**: Manufacturing by supplier
- **Delivery by supplier**: Delivery by supplier

### CNC Dimensioning System

- **Supplier search (tech requirements, budgetary offers)**: Supplier search (tech requirements, budgetary offers)
- **Investment proposal (VTT template)**: Investment proposal (VTT template)
- **Competitive bid process (notice, evaluation, decision)**: Competitive bid process (notice, evaluation, decision)
- **Procurement Decision and placing of order**: Procurement Decision and placing of order
- **Manufacturing by supplier**: Manufacturing by supplier
- **Delivery by supplier**: Delivery by supplier

### Standard-sized Instrumented Impact Hammer

- **Supplier search (tech requirements, budgetary offers)**: Supplier search (tech requirements, budgetary offers)
- **Investment proposal (VTT template)**: Investment proposal (VTT template)
- **Competitive bid process (notice, evaluation, decision)**: Competitive bid process (notice, evaluation, decision)
- **Procurement Decision and placing of order**: Procurement Decision and placing of order
- **Manufacturing by supplier**: Manufacturing by supplier
- **Delivery by supplier**: Delivery by supplier

### Microscopy devices

### TASK 2.2 Microscopy devices

#### REMOVED FROM RADLAB 2017 PER PROPOSAL EVALUATION

### TASK 2.3 Nuclear waste management devices (Laevonen)

#### NOT FUNDED BY SAFIR, BUT FUNDED BY KYT

- **Liquid Scintillation Counter**: Liquid Scintillation Counter
- **Supplier search (tech requirements, budgetary offers)**: Supplier search (tech requirements, budgetary offers)
- **Investment proposal (VTT template)**: Investment proposal (VTT template)
- **Competitive bid process (notice, evaluation, decision)**: Competitive bid process (notice, evaluation, decision)
- **Procurement Decision and placing of order**: Procurement Decision and placing of order
- **Manufacturing by supplier**: Manufacturing by supplier
- **Delivery by supplier**: Delivery by supplier
- **Installation VTT inspection**: Installation VTT inspection

#### Bentonite Aineenkuulutuslaite

- **Investment proposal (VTT template)**: Investment proposal (VTT template)
- **Supplier search (tech requirements, budgetary offers)**: Supplier search (tech requirements, budgetary offers)
- **Sollicitation of final offer**: Sollicitation of final offer
- **Procurement Decision and placing of order**: Procurement Decision and placing of order
- **Manufacturing by supplier**: Manufacturing by supplier
- **Delivery by supplier**: Delivery by supplier

#### Other Investment Aid investments managed in WP2.

- **Technical support services equipping**: Technical support services equipping
  - **Supplier search (tech requirements, budgetary offers)**: Supplier search (tech requirements, budgetary offers)
  - **Investment proposal (VTT template)**: Investment proposal (VTT template)
  - **Competitive bid process (notice, evaluation, decision)**: Competitive bid process (notice, evaluation, decision)
  - **Procurement Decision and placing of order**: Procurement Decision and placing of order
  - **Manufacturing by supplier**: Manufacturing by supplier
  - **Delivery by supplier**: Delivery by supplier
  - **Required accessories selection and purchase**: Required accessories selection and purchase
### Work Package 3 Work and Investments

<table>
<thead>
<tr>
<th>TASK 3.1 Autoclaves</th>
<th>2016</th>
<th>2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vesipiiri autoklaaveille</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Assessment of needs (in lieu of investing in fully new one)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Invitation for tenders</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Procurement of needed components</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Component replacement and high-pressure pump maint.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Basic functionality testing in old facilities</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Transport to CNS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Basic functionality testing</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Initiation of proving trials (within EU-SOTERIA project)</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Autoklaavi 1 (replacement of old 2 x SSRT)</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Assessment of needs</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Invitation for tenders</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Procurement</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Installation in CNS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Basic functionality testing</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Autoklaavi 2 (upgrade of 1 x SSRT)</strong></td>
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<tr>
<td>Assessment of needs</td>
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<td></td>
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<tr>
<td>Invitation for tenders</td>
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<tr>
<td>Procurement of needed components</td>
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<td></td>
</tr>
<tr>
<td>Component replacement</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Preparation for transport to CNS (disassembly, packing)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Transport to CNS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Installation to CNS hot autoclave lab</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Development of manipulation method for hot specimens</td>
<td></td>
<td></td>
</tr>
<tr>
<td>M3.1.1 Acquisition of last &quot;hot&quot; autoclave refurbishment components</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Initiation of proving trials (within EU-SOTERIA project)</td>
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<td></td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>TASK 3.2 Nuclearization of equipment</th>
<th>2016</th>
<th>2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>Local shielding &amp; containments (1 MTS, 1 Impact)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Conceptual design of gamma shielding &amp; containment</td>
<td></td>
<td></td>
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<tr>
<td>Engineering design of shielding &amp; containment</td>
<td></td>
<td></td>
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<tr>
<td>Investment proposals for outsourced components</td>
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<td></td>
</tr>
<tr>
<td>Manufacturing drawings for outsourced components</td>
<td></td>
<td></td>
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<tr>
<td>Procurement process for outsourced work</td>
<td></td>
<td></td>
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<tr>
<td>Moving of locally-shielded devices from OK3</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Manufacturing of in-house and outsourced components</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Installation of locally shielded equipment with shielding</td>
<td></td>
<td></td>
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<tr>
<td>Functionality testing of locally shielded facilities</td>
<td></td>
<td></td>
</tr>
<tr>
<td>D3.2.1 Descriptive report of locally shielded equipment installations</td>
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</table>

### EBW- nuclearization

<table>
<thead>
<tr>
<th>EBW- nuclearization</th>
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<th>2017</th>
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</thead>
<tbody>
<tr>
<td>Design of nuclearization features with EBW supplier</td>
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<td></td>
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<tr>
<td>Solicitation of final offer</td>
<td></td>
<td></td>
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<tr>
<td>Procurement Decision and placing of order</td>
<td></td>
<td></td>
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<tr>
<td>Planning and preparation of moving</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Manufacturing by supplier</td>
<td></td>
<td></td>
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<tr>
<td>Moving of equipment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Delivery and nuclearization by supplier</td>
<td></td>
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</tr>
<tr>
<td>M3.2.1 In-cell installation of nuclearized EBW device</td>
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<td></td>
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</tbody>
</table>

### EDM- nuclearization and installation.

<table>
<thead>
<tr>
<th>EDM- nuclearization and installation.</th>
<th>2016</th>
<th>2017</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design of nuclearization EDM features</td>
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<tr>
<td>Planning and preparation of moving</td>
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<td></td>
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<tr>
<td>Manufacturing by supplier</td>
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<tr>
<td>Moving of equipment</td>
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<tr>
<td>Pre-installation preparations</td>
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<tr>
<td>M3.2.2 In-cell installation of nuclearized EDM device</td>
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<td></td>
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<tr>
<td>Construction of water circuit for installed EDM</td>
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</tbody>
</table>

To reflect low priority in proposal evaluation, the installation and testing will not be carried out as part of RADLAB.

**Note:**
- VTT Investment proposal deadline
- VTT Fiscal year begins
- Earliest release of OK3 devices
- Deadline for A-lab permit
- Deadline for A-lab permit
- Deadline for A-lab permit
- Deadline for A-lab permit
- Deadline for A-lab permit
- Deadline for A-lab permit
## Work Package 4 Work and Investments

### TASK 4.1 Waste handling installation

<table>
<thead>
<tr>
<th>Year</th>
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<th>Mar</th>
<th>Apr</th>
<th>May</th>
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</tbody>
</table>

- Nestemäisten jätteiden käsittelylaitteisto
- Conceptual design and technical description
- Engineering design procurement process
- Design work
- Component procurement
- Delivery and installation
- Procedure development and implementation

### TASK 4.2 Storage and Logistics

<table>
<thead>
<tr>
<th>Year</th>
<th>Jan</th>
<th>Feb</th>
<th>Mar</th>
<th>Apr</th>
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</table>

- Radioaktiivisten materiaalien tietokantajärjestelmä
- Technical requirements
- Conceptual Design and technical description
- Procurement process
- Realization of system with subcontractor
- Functionality test of specimen storage database

### D4.1.1: Descriptive report of waste handling installation

<table>
<thead>
<tr>
<th>Year</th>
<th>Jan</th>
<th>Feb</th>
<th>Mar</th>
<th>Apr</th>
<th>May</th>
<th>Jun</th>
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</table>

- Kattonostuun tynnityventilo
- Assessment of installation conditions
- Procurement process
- Manufacturing by supplier
- Delivery and functionality testing
- Procedure development and implementation
- Component procurement
- Delivery and installation

### D4.2.1: Engineering and fabrication contract for specimen storage system

<table>
<thead>
<tr>
<th>Year</th>
<th>Jan</th>
<th>Feb</th>
<th>Mar</th>
<th>Apr</th>
<th>May</th>
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</tbody>
</table>

- Kuljetusaista, iso
- Supplier search (tech requirements, budgetary offers)
- Investment proposal (VTT template)
- Competitive bid process (notice, evaluation, decision)
- Procurement Decision and placing of order
- Delivery by supplier
### Work packages and Tasks

<table>
<thead>
<tr>
<th>Expenses</th>
<th>Financing</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Volume</strong></td>
<td><strong>Personnel</strong></td>
</tr>
<tr>
<td>WP1 - Hot Cells</td>
<td></td>
</tr>
<tr>
<td>T1.1 Hot cell installation with ITD</td>
<td>12.0</td>
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<tr>
<td>T1.2 Training and HOTLAB</td>
<td>4.0</td>
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<tr>
<td>WP2 - Hot Lab equipment procurement</td>
<td>3.0</td>
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<tr>
<td>T2.1 Mechanical test devices</td>
<td>2.0</td>
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<tr>
<td>T2.2 Metallography (Removed)</td>
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<tr>
<td>T2.3 NWM devices (No SAFIR)</td>
<td>1.0</td>
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<tr>
<td>WP3 - Research equipment</td>
<td>8.3</td>
</tr>
<tr>
<td>T3.1 Hot autoclave facilities</td>
<td>2.0</td>
</tr>
<tr>
<td>T3.2 Nuclearization and in-cell devices</td>
<td>5.3</td>
</tr>
<tr>
<td>T3.3 NWM auxiliary devices (No SAFIR)</td>
<td>1.0</td>
</tr>
<tr>
<td>WP4 - Supporting facilities</td>
<td>12.5</td>
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<tr>
<td>T4.1 Laboratory waste handling systems</td>
<td>5.0</td>
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<tr>
<td>T4.2 Specimen storage systems</td>
<td>7.5</td>
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<tr>
<td>WP5 - VTT CNS</td>
<td>7.0</td>
</tr>
<tr>
<td>T5.1 NWM laboratory and ICP-MS utilization</td>
<td>4.0</td>
</tr>
<tr>
<td>T5.2 A-class facilities commissioning and project mgmt</td>
<td>3.0</td>
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<tr>
<td><strong>TOTAL</strong></td>
<td>46.8</td>
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</tbody>
</table>

**Comments:**
- Travel allocations for T1.1 are for two people to attend the FAT#5 at ITD (Germany)
- Travel allocations for T1.2 are for two people to participate in the HOTLAB conference in Japan
- Travel allocations for T5.1 are for up to 4 people to participate in ICP-MS training and conferences
- Other travel allocations are mainly for demonstrations of equipment, equipment FATs, etc.
- The expenses in the Other column are the mandatory research infrastructure surcharge, which covers materials, supplies and external services

**Instructions:**
- The equations must be modified according to the number of WP’s and tasks. The numbers related to tasks should be summed to the WP lines (above).
- Also the equations in the TOTAL line need to be modified.
- The "Check sum" column should give the same numbers as the TOTAL column (it is outside the printing area).
- Rental of equipment can be included in the "Other" column and explained on the "Comments" line.
- Financing column headers can be edited as needed and unnecessary columns can be removed (VYR column must be included also in the case of zero financing).
Appendix 2

The Management Board, the Steering Groups and the Reference Groups in 2017
<table>
<thead>
<tr>
<th>Organisation</th>
<th>Member</th>
<th>Vice member</th>
</tr>
</thead>
<tbody>
<tr>
<td>STUK</td>
<td>Marja-Leena Järvinen (Chair)</td>
<td>Tomi Routamo</td>
</tr>
<tr>
<td>STUK</td>
<td>Tomi Routamo (Vice chair)</td>
<td>Nina Lahtinen</td>
</tr>
<tr>
<td>Aalto</td>
<td>Filip Tuomisto</td>
<td>Eila Järvenpää</td>
</tr>
<tr>
<td>Fennovoima</td>
<td>Hanna Virlander</td>
<td>Vesa Ruuska</td>
</tr>
<tr>
<td>Fortum</td>
<td>Kristiina Söderholm</td>
<td>Matti Kattainen</td>
</tr>
<tr>
<td>LUT</td>
<td>Juhani Hyvärinen</td>
<td>Heikki Purhonen</td>
</tr>
<tr>
<td>MEAE</td>
<td>Jorma Aurela</td>
<td>Linda Kumpula</td>
</tr>
<tr>
<td>SSM</td>
<td>Nils Sandberg (on leave)</td>
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<tr>
<td>Tekes</td>
<td>Arto Kotipelto</td>
<td>Reijo Munther</td>
</tr>
<tr>
<td>TVO</td>
<td>Antti Tarkiainen</td>
<td>N/A</td>
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<tr>
<td>VTT</td>
<td>Eija Karita Puska</td>
<td>Petri Kinnunen</td>
</tr>
<tr>
<td>SAFIR2018 (Secretary)</td>
<td>Jari Hämäläinen</td>
<td>Vesa Suolanen</td>
</tr>
</tbody>
</table>
SAFIR2018 Steering Groups:

**SG1 – Plant safety and systems engineering**

<table>
<thead>
<tr>
<th>Organisation</th>
<th>Member</th>
<th>Vice member</th>
</tr>
</thead>
<tbody>
<tr>
<td>STUK</td>
<td>Tomi Routamo (Chair)</td>
<td>Eero Virtanen</td>
</tr>
<tr>
<td>Fennovoima</td>
<td>Pekka Viitanen</td>
<td>Juho Helander</td>
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<tr>
<td>Fortum</td>
<td>Eero Vesaoja</td>
<td>Maria Vuokko</td>
</tr>
<tr>
<td>TVO</td>
<td>Jari Pesonen (Vice chair)</td>
<td>Mikko Lemmetty</td>
</tr>
<tr>
<td>SAFIR2018 (Secretary)</td>
<td>Jari Hämäläinen</td>
<td>Vesa Suolanen</td>
</tr>
</tbody>
</table>

**SG2 – Reactor safety**

<table>
<thead>
<tr>
<th>Organisation</th>
<th>Member</th>
<th>Vice member</th>
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<td>Jukka Rintala</td>
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<td>SAFIR2018 (Secretary)</td>
<td>Jari Hämäläinen</td>
<td>Vesa Suolanen</td>
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**SG3 – Structural safety and materials**

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SAFIR2018 – Reference Groups and Projects:

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<td><strong>RG1 Automation, organisation and human factors</strong></td>
<td>CORE (SG1), ESSI (SG1), MAPS (SG1), SAUNA (SG1)</td>
<td>SG1 area</td>
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<tr>
<td>(Automaatio, organisaatio ja inhimilliset tekijät)</td>
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<td><strong>RG2 Severe accidents and risk analysis</strong></td>
<td>EXWE (SG1), FIRED (SG3), ERNEST (SG3) CASA (SG2), CATFIS (SG2), GENXFIN (SG1), PRAMEA (SG1)</td>
<td>SG1, SG2 and SG3 areas</td>
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<td>(Vakavat onnettomuudet ja riskitutkimus)</td>
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<td><strong>RG3 Reactor and fuel</strong></td>
<td>KATVE (SG2), MONSOON (SG2), PANCHO (SG2), SADE (SG2), USVA (SG2)</td>
<td>SG2 area</td>
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<td>(Reaktori ja polttoaine)</td>
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<td><strong>RG4 Thermal hydraulics</strong></td>
<td>COVA (SG2), INSTAB (SG2), INTEGRA (SG2), NURESA (SG2)</td>
<td>SG2 area</td>
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<td>(Termohydraulikka)</td>
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<td><strong>RG5 Structural integrity</strong></td>
<td>COMRADE (SG3), FOUND (SG3), LOST (SG3), MOCCA (SG3), THELMA (SG3), WANDA (SG3)</td>
<td>SG3 area</td>
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<td>(Rakenteellinen eheys)</td>
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<tr>
<td><strong>RG6 Research infrastructure</strong></td>
<td>INFRAL, JHR, RADLAB</td>
<td>RG6 area</td>
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<td>(Tutkimusinfrastruktuuri)</td>
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### RG1 – Automation, organisation and human factors

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<td>Janne Peltonen (Vice chair), Anna Aspelund</td>
<td>Topi Tahvonen</td>
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<td>Fortum</td>
<td>Juha Lamminen, Jussi Lahtinen, Maria Vuokko</td>
<td>Ville Nurmilaukas</td>
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<td>LUT</td>
<td>Anne Jordan, Eetu Kotro</td>
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<td>STUK</td>
<td>Mika Johansson, Pia Oedewald, Paula Savioja</td>
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<td>Mauri Viitasalo (Chair), Petri Koistinen</td>
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### RG2 – Severe accidents and risk analysis

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<td>Jani Laine</td>
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### RG3 – Reactor and fuel

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### RG4 – Thermal hydraulics

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## RG5 – Structural integrity

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## RG6 – Research infrastructure

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